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Safety assessment for a KBS-3H spent nuclear fuel repository at Olkiluoto

Summary report

Paul Smith, Fiona Neall, Margit Snellman, Barbara Pastina, Henrik Nordman, Lawrence Johnson, Thomas Hjerpe

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Svensk Kärnbränslehantering AB
Swedish Nuclear Fuel
and Waste Management Co
Box 250, SE-101 24 Stockholm
Tel +46 8 459 84 00

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Abstract

The KBS-3 method, based on multiple barriers, is the proposed spent fuel disposal method both in Sweden and Finland. KBS-3H and KBS-3V are the two design alternatives of the KBS-3 spent fuel disposal method. Posiva and SKB have conducted a joint research, demonstration and development (RD&D) programme in 2002–2007 with the overall aim of establishing whether KBS-3H represents a feasible alternative to the reference alternative KBS-3V. The overall objectives of the present phase covering the period 2004–2007 have been to demonstrate that the horizontal deposition alternative is technically feasible and to demonstrate that it fulfils the same long-term safety requirements as KBS-3V. The safety studies conducted as part of this programme include a safety assessment of a preliminary design of a KBS-3H repository for spent nuclear fuel located about 400 m underground at the Olkiluoto site, which is the proposed site for a spent fuel repository in Finland. This safety assessment is summarised in the present report.

The scientific basis of the safety assessment includes around 30 years of scientific R&D and technical development in the Swedish and Finnish KBS-3V programmes. Much of this scientific basis is directly applicable to KBS-3H. This has allowed the KBS-3H safety studies to focus on those issues that are unique to this design alternative, identified in a systematic "difference analysis" of KBS-3H and KBS-3V. This difference analysis has shown that the key differences in the evolution and performance of KBS-3H and KBS-3V relate mainly to the engineered barrier system and to the impact of local variations in the rate of groundwater inflow on buffer saturation along the KBS-3H deposition drifts. No features or processes specific to KBS-3H have been identified that could lead to a loss or substantial degradation of the safety functions of the engineered barriers over a million year time frame. Radionuclide release from the repository near field in the event of canister failure may be affected by perturbations due, for example, to the presence of steel components external to the canister in the present KBS-3H reference design. These will corrode over time and interact with the bentonite buffer, affecting its transport properties. In spite of such perturbations, calculated releases are limited and comply with Finnish regulatory criteria in the cases considered.

The present safety assessment has some important limitations. In particular, the analysis of a limited range of assessment cases with highly simplified models, especially of the geosphere, is not considered sufficient to test whether the current KBS-3H design at the Olkiluoto site satisfies all relevant regulatory guidelines. Further limitations are that the feasibility of implementing the current reference design has been assumed, even though several design issues remain to be addressed. Furthermore, only single canister failure cases have been considered. Nevertheless, it can be concluded, based on the present safety assessment, that the KBS-3H design alternative offers potential for the full demonstration of safety for a repository at Olkiluoto site and for the demonstration that it fulfills the same long-term safety requirements as KBS-3V. Remaining critical scientific and design issues are highlighted in this report. These include the further development of the DAWE (Drainage, Artifical Watering and air Evacuation) design alternative to avoid uncertainties associated with the buffer saturation process, as well as studies of iron/bentonite interaction and the possible use of materials such as titanium in place of steel for some system components.

This report has also been published as a Posiva report, POSIVA 2007-06.

Sammanfattning

KBS-3 metoden, som bygger på multibarriärprincipen, har valts för slutdeponering av använt kärnbränsle i Finland och utgör planeringsförutsättningen i Sverige. KBS-3H och KBS-3V är två alternativa utformningar av KBS-3 metoden. Posiva och SKB har utfört ett gemensamt forsknings-, demonstrations- och utvecklingsprogram (FUD) under 2002–2007 med det övergripande målet att utvärdera om KBS-3H kan utgöra ett alternativ till referensalternativet KBS-3V. Det övergripande målet för denna fas, omfattande perioden 2004–2007, har varit att demonstrera att det horisontella deponeringsalternativet är tekniskt genomförbart och att demonstrera att det uppfyller samma krav på långtidssäkerhet som KBS-3V. Säkerhetsstudierna, som utförts som en del av detta program, omfattar en säkerhetsanalys av en preliminär utformning av KBS-3H för ett slutförvar för använt kärnbränsle lokaliserat på 400 m djup i Olkiluoto, som är den föreslagna platsen för ett slutförvar för använt kärnbränsle i Finland. Denna säkerhetsanalys har sammanfattats i denna rapport.

Den vetenskapliga grunden för säkerhetsanalysen baserar sig på 30 år av vetenskaplig forskning och teknisk utveckling inom KBS-3V-programmet i Sverige och Finland. Mycket av denna kunskap kan direkt tillämpas på KBS-3H. Detta har möjliggjort att KBS-3H:s säkerhetsstudier kunnat fokuseras på de frågor som är unika för detta alternativ. Frågorna har identifierats i en systematisk differensanalys mellan KBS-3H och KBS-3V. Denna analys har påvisat att den huvudsakliga skillnaden i utvecklingen av KBS-3H och KBS-3V mest är relaterad till hur de tekniska barriärerna utvecklas med tiden och till hur den lokala variationen i vatteninflöde i den långa deponeringstunneln för KBS-3H inverkar på svällningen av bufferten. Inga KBS-3H-specifika egenskaper eller processer har identifierats, som skulle kunna leda till en betydande förlust av säkerhetsfunktionerna hos de tekniska barriärerna i ett tidsperspektiv på en miljon år. Radionuklidernas transport från slutförvarets närområde vid en kapselskada kan påverkas av störningar i närområdet på grund av närvaron av stålkomponenter utanför kapseln i den nuvarande KBS-3H-utformningen. Dessa stålkomponenter korroderar med tiden och växelverkar med bentonitbufferten. Korrosionsprodukterna inverkar på transportegenskaperna i närområdet. Även med beaktande av dessa störningar är de beräknade utsläppen begränsade i omfattning och ligger inom ramen för de finska myndighetskriterier i de fall som analyserats.

Den nuvarande säkerhetsanalysen har några betydande begränsningar. Ett fåtal analyserade fall med mycket förenklade modeller, speciellt för geosfären, gör att analysen är otillräcklig för att bedöma om den nuvarande KBS-3H-utformningen av ett slutförvar i Olkiluoto uppfyller myndigheternas samtliga föreskrifter. Ytterligare begränsningar är relaterade till antagandet att den nuvarande utformningen är genomförbar även om flera utformningsfrågor ännu måste klarställas. Dessutom har endast fallet med en skadad kapsel hanterats i analysen. Trots dessa begränsningar är slutsatsen att KBS-3H-alternativet har förutsättningar för säker slutförvaring i Olkiluoto och för uppfyllande av samma krav på långtidssäkerheten som KBS-3V. Kvarstående kritiska vetenskapliga och tekniska frågor har tagits fram i denna rapport. Dessa inkluderar bl a vidareutveckling av utformningsvarianten DAWE (Drainage, Artificial Watering and air Evacuation) för att undvika osäkerheter som är kopplade till buffertens mättnadsprocesser, och ytterligare studier av växelverkan mellan järn-bentonit och möjlig användning av titan i stället för stål i vissa systemkomponenter.

Denna rapport finns även tryckt i Posivas rapportserie POSIVA 2007-06.

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Foreword

This study was coordinated by Margit Snellman from Saanio & Riekkola Oy on behalf of Posiva Oy. The progress of the study was supervised by a KBS-3H Review Group consisting of Aimo Hautojärvi (Posiva), Jukka-Pekka Salo (Posiva), Marjut Vähänen (Posiva), Barbara Pastina (Saanio & Riekkola Oy), Margit Snellman (Saanio & Riekkola Oy), Jorma Autio (Saanio & Riekkola Oy), Stig Pettersson (SKB), Erik Thurner (SKB), Börje Torstenfelt (Swedpower), Lennart Börgesson (Clay Technology) and Lawrence Johnson (Nagra). The present Summary Report was largely written by Paul Smith (Safety Assessment Management Ltd.), with contributions from Fiona Neall (Neall Consulting Ltd.), Margit Snellman (Saanio & Riekkola Oy), Barbara Pastina (Saanio & Riekkola Oy), Henrik Nordman (VTT), Lawrence Johnson (Nagra) and Thomas Hjerpe (Saanio & Riekkola Oy).

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Executive summary

Background and aims

The KBS-3 method, based on multiple barriers, is the proposed spent fuel disposal method both in Sweden and Finland. KBS-3H and KBS-3V are the two design alternatives of the KBS-3 spent fuel disposal method. Posiva and SKB have conducted a joint research, demonstration and development (RD&D) programme in 2002–2007 with the overall aim of establishing whether KBS-3H represents a feasible alternative to KBS-3V. The overall objectives of the present phase covering the period 2004–2007 have been to demonstrate that the horizontal deposition alternative is technically feasible and to demonstrate that it fulfils the same long-term safety requirements as the reference design KBS-3V. The safety studies conducted as part of this programme include a safety assessment of a preliminary design of a KBS-3H repository for spent nuclear fuel located about 400 m underground at the Olkiluoto site, which is the proposed site for a spent fuel repository in Finland. This safety assessment is summarised in the present report.

In KBS-3H the spent nuclear fuel is encapsulated in copper canisters with cast iron inserts, and the canisters are emplaced horizontally along deposition drifts. This contrasts with KBS-3V, in which canisters are emplaced vertically in individual deposition holes. KBS-3V is currently the reference design alternative in both Finland and Sweden.

Specific high-level questions addressed by the safety assessment and by the wider safety studies are:

- Are there safety issues specific to KBS-3H with the potential to lead to unacceptable[1] radiological consequences?
- Is KBS-3H promising at a site with the broad characteristics of Olkiluoto from the long-term safety point of view?

The safety studies are, however, limited in scope and do not currently address the questions:

- Is KBS-3H more or less favourable than KBS-3V from a long-term safety point of view?
- Does the specific realisation of the KBS-3H repository design considered in the safety studies satisfy all relevant regulatory guidelines?

Safety concept and safety functions

The safety concept for both KBS-3H and KBS-3V is based on long-term isolation of spent fuel and containment of radionuclides. In KBS-3H as in KBS-3V, the copper canister, the bentonite clay buffer surrounding the canister and the host rock are the main system components that together ensure long-term safety. In order to assess the performance and safety of a repository, it is necessary to determine the conditions under which the identified safety functions of these components will operate as intended, and the conditions under which they will fail, or operate with reduced effectiveness. Following the methodology adopted in the Swedish SR-Can safety assessment, one or more safety function indicators are assigned to each of the safety functions. If the safety function indicators fulfil certain criteria, then the safety functions can be assumed to be provided. If, however, plausible situations can be identified where the criteria for one or more safety function indicators are not fulfilled, then the consequences of loss or degraded performance of the corresponding safety function must be evaluated as part of the safety assessment. The majority of the criteria used in the present safety assessment are taken from SR-Can.

[1] According to the Finnish regulatory requirements in YVL 8.4.

Design, initial conditions, processes and evolution

The safety assessment includes a description of the initial conditions within and around a KBS-3H deposition drift, based largely on the design specifications of the repository, on the Olkiluoto site reports, and on a repository layout report. It also includes a description of processes that may occur within and around the repository over time, and a description of the evolution of the repository in successive time frames, including a description of the main uncertainties affecting this evolution. The features, events and processes (FEPs) considered in KBS-3H process descriptions have been checked for completeness by auditing them against the SR-Can Process Reports and FEP database, and also against the international FEP database maintained by the OECD/NEA. The descriptions of processes and repository evolution over successive time frames provide the basis for the identification of evolution scenarios, an assessment of canister longevity and the analysis of radionuclide release and transport in the event of canister failure.

These descriptions assume a preliminary reference repository design, termed the Basic Design, although a number of design variants are currently under consideration. At the time of selection of the reference design for long-term safety studies, no major differences between the selected reference design Basic design and the design alternative, termed DAWE (Drainage Artificial Watering and air Evacuation), had been identified that were relevant to long-term safety. In the Basic Design, each canister, with a surrounding partly saturated layer of bentonite clay, is placed in a perforated steel cylinder prior to emplacement and the entire assembly is called the supercontainer. The supercontainers are supported by steel feet and are positioned along parallel, 100–300 m long deposition drifts. The drift separation is 25 m and the canister pitch is 11 m in Posiva's layout for the Olkiluoto site. Adjacent supercontainers in the drift are separated from each other by bentonite distance blocks. The distance blocks are kept in place by fixing rings. The bentonite originally inside the supercontainers and the bentonite distance blocks jointly make up the buffer. The drifts are sealed using steel-reinforced low-pH concrete drift end plugs. Groundwater control measures to reduce inflow to the drifts will be implemented for reasons of engineering practicality and operational safety, and to limit the possibility of phenomena that could be detrimental to long-term safety, such as piping and erosion. These measures include the use of steel compartment plugs to seal off high inflow drift sections, thus dividing the drift into isolated compartments.

Difference analyses approach

The scientific basis of the safety assessment includes around 30 years of scientific R&D and technical development in the Swedish and Finnish KBS-3V programmes. Much of this scientific basis is directly applicable to KBS-3H. This has allowed KBS-3H safety studies and the safety assessment to focus on those issues that are unique to KBS-3H, identified in a systematic "difference analysis" of KBS-3H and KBS-3V. In both alternatives, the system evolves from its initial state through an early, transient phase towards a quasi-steady state, in which key safety-relevant physical and chemical characteristics (e.g. temperature, buffer density and swelling pressure) are subject to much slower changes than in the transient phase. Long-term safety assessment starts from the time of emplacement of the first canisters in the repository. This is also the starting point for the early evolution of the system. The end-point of early evolution is not well defined; many of the transient processes that occur during this period do not suddenly cease, but rather gradually diminish over time. Nevertheless, two key transient processes – heat dissipation from the spent fuel and saturation of the repository near field – may take up to several thousands of years (or even longer in the case of saturation of the tightest sections) and this may be taken as the rough duration of the early evolution period. The difference analysis has shown that most of the differences between KBS-3H and KBS-3V relate to internal processes involving KBS-3H-specific components, such as the supercontainer and other structural components, and variations in hydraulic conditions in KBS-3H deposition drifts and their immediate environment (supercontainer-buffer-rock interface, near-field rock, drift end plugs). Many of these differences affect the early, transient phase of repository evolution, when significant mass and energy fluxes will occur as a result of the various gradients created by repository construction and emplacement of spent fuel, although some differences also occur at later times. For example, the radionuclide transport paths from a failed canister are affected by the differences in the geometry and backfilling of the KBS-3H deposition drifts compared with the KBS-3V deposition tunnels.

Key safety issues and evolution scenarios

The descriptions of processes and system evolution for a KBS-3H repository, and the difference analysis with KBS-3V, indicates that most safety issues are common to the two alternatives, but that there are also differences. Often, these are differences in the significance to, or potential impact of, an issue to each of the alternatives. Key safety issues that are judged to have a different significance to, or potential impact on, KBS-3H compared with KBS-3V, concern mainly the early, transient evolution of the repository. They include

(i) piping and erosion during repository operations and drift saturation,

(ii) steel components external to the canister, their corrosion products and their impact on mass transport,

(iii) impact of gas from the corrosion of steel components external to the canister,

(iv) interactions involving leachates from cementitious components,

(v) thermally-induced rock spalling,

(vi) expulsion of water and dissolved radionuclides from a defective canister interior by gas.

Issue (i) is a critical issue for repository design as well as safety. Design measures must be taken to limit the possibility of significant piping and erosion during early evolution of the buffer. At the current design stage, however, the possibility of some degree of piping and erosion during buffer saturation cannot be completely eliminated, with potential consequences for canister corrosion (via its impact on the transport of sulphide from the groundwater through the buffer to the canister surface) and radionuclide transport in the event of canister failure. In the absence of other more significant perturbations to mass transport in the buffer or at the buffer/rock interface (see below), scoping calculations indicate that limited piping and erosion will not lead to canister failure by copper corrosion within a million year time frame. This is due to the limited sulphide concentration in the groundwater (although the variability and evolution of groundwater sulphide concentration with time is an issue for future study) and to the slow transport of sulphide through the buffer, which is expected to remain diffusion dominated even following the local erosion of up to a few hundred kilograms of bentonite, due to the swelling and homogenisation of this material over time. The consequences in terms of radionuclide releases to the geosphere and biosphere in the event of canister failure have been evaluated by means of radionuclide release and transport calculations as part of the safety assessment.

The most significant potential impact of issues (ii), (iv) and (v) is on mass transfer at the buffer/rock interface (the other issues and associated processes may also have some limited effects on the interface). Perturbation of mass transfer across the interface may again affect canister corrosion via its effect on the transfer of groundwater sulphide to the canister surface, and affect radionuclide release to the geosphere and biosphere in the event of canister failure. Scoping calculations indicate that the presence of a perturbed buffer/rock interface has the potential to lead to canister failure by copper corrosion within a million year time frame in the case of canisters located near to more transmissive fractures, particularly if the sulphide concentration at the buffer/rock interface is significantly increased, e.g. by microbial activity. However, no canister failures are expected before about 100,000 years. Failure of a single canister by corrosion at 100,000 years is considered in radionuclide release and transport assessment cases addressing this particular canister failure mode. The presence of a hydraulically conductive zone at the buffer/rock interface due to (ii), (iv) or (v) could also perturb radionuclide release from the buffer to the geosphere in the event of canister failure, irrespective of the failure mode, and this possibility is also considered in radionuclide release and transport calculations.

Issue (iii), hydrogen gas generated by the corrosion of steel in the supercontainers and other steel components external to the canisters may affect the saturation of the repository. In the tightest drift sections, repository-generated gas may hinder or prevent altogether the saturation of the buffer until gas generation by steel corrosion ceases and gas pressure falls, which may take up to tens of thousands of years. The impact on radionuclide release to the geosphere has not as yet been quantified, but releases are expected to be no more than in the case of a fully saturated buffer, and may be somewhat reduced. Hydrogen generation may also affect the corrosion of the canisters, via its effect on the microbial reduction of sulphate to sulphide. Scoping calculations that include this effect show that, even in the case of a pessimistically modelled perturbed buffer/rock interface, an overall canister lifetime of several hundred thousand years is expected. However, the impact of the hydrogen on bentonite porewater chemistry has yet to be evaluated. Finally, as gas rises through the fracture network within the host rock, it may carry water with it. Any perturbation to groundwater flow in the geosphere due to this process is, however, likely to have largely ceased by the time most radionuclides are released from failed canisters due to the limited duration of gas generation by corrosion (a few thousand years), except possibly in the tightest drift sections where the dissipation of generated gas is very slow, where in any case groundwater flow is virtually zero.

Issue (vi), expulsion of water and dissolved radionuclides from an initially penetrated canister by gas, is a possibility if the defect is located on the lower side of the horizontally orientated canister. In this case, it is possible that gas generated principally by corrosion of the insert will become trapped above water lying in the lowest part of canister, and gas pressure will build up until it is sufficient to expel the water and dissolved radionuclides into the buffer. Scoping calculations indicate that the more likely situation is that water entering the canister will be completely consumed by corrosion of the cast iron insert, and there will be no gas-induced displacement of contaminated water through the defect into the saturated bentonite. The possibility of expulsion of contaminated water by gas cannot, however, be completely excluded, and its impact on radionuclide release and transport is addressed in a radionuclide release and transport assessment case.

In addition to these key safety issues that are judged to have a different significance to, or potential impact on, KBS-3H compared with KBS-3V; there are a number of other key issues that are judged to have similar significance or potential impact. These issues, which relate to longer-term evolution, include:

(i) buffer freezing, which is not expected to occur according to present knowledge based on past glaciations, although the possibility that conditions at Olkiluoto could in the future differ significantly compared with those during the past glaciations requires further study,

(ii) canister failure due to isostatic load, which can be ruled out in the case of a KBS-3H repository at Olkiluoto (as it was in SR-Can),

(iii) oxygen penetration to repository depth in association with future glacial cycles, the possibility of which is excluded in the present safety assessment based on the recent interpretation of hydrogeochemical data from the Olkiluoto site, although more information to support this tentative finding will be required in future studies,

(iv) canister failure due to rock shear in the event of a large earthquake, which cannot currently be excluded and is a failure mode that is considered in radionuclide release and transport assessment cases,

(v) loss of buffer from exposure to glacial meltwater ("chemical erosion"), which also cannot be ruled out based on current, limited understanding of this process – canister failure by corrosion following the establishment of advective conditions in the buffer is thus also a canister failure mode that is considered in radionuclide release and transport assessment cases,

(vi) implications of a prolonged period of temperate climate ("greenhouse effect"), which is concluded to be generally beneficial to safety (as it was in SR-Can), since the long-term processes that are potentially the most detrimental to repository safety are related to glacial conditions.

By considering the potential impact of the key safety issues – those with different significance to, or potential impact on, KBS-3H compared with KBS-3V as well as others that are common to KBS-3H and KBS-3V – on the safety functions of the repository, and taking into account the processes affecting the evolution of the repository and site, various possible evolution scenarios have been identified (Figure 1).

The Base Scenario assumes (as required by Finnish regulations) that the performance targets defined for each barrier are met. This is interpreted as meaning that each barrier fulfils the safety functions assigned to it in the safety concept for a period extending to a million years or more. There are, however, scenarios whereby one or more canister failures lead to radionuclide release and transport and exposure of humans and other biota to released radionuclides, in a one million year time frame. These are initiated, in the first place, by:

- the presence of an initial, penetrating defect in one or more of the canisters,
- perturbations to the buffer and buffer/rock interface, giving rise to an increased rate of transport of sulphide from the geosphere to the canister surface and an increased canister corrosion rate,
- penetration of dilute glacial meltwater to repository depth, giving rise to chemical erosion of the buffer an increased rate of transport of sulphide from the geosphere to the canister surface and an increased canister corrosion rate,
- rock shear movements of sufficient magnitude to give rise to shear failure of the canisters.

Due to limitations in the understanding of relevant processes, only in the case of canister failure by rock shear following a large earthquake is an estimate made of the likelihood or rate of canister failure. According to Finnish regulations, the importance to safety of a substantial rock movement occurring in the environs of the repository should be considered. Scoping calculations give an expectation value of 16 out of the total number of 3,000 canisters as the number of canisters in the repository that could potentially be damaged by rock shear in the event of a large earthquake. Such calculations are based on a statistical description of the network of fractures at the site and significant uncertainties have been identified that could lead to either an underestimate or an overestimate of the actual likelihood of canister damage. Major seismic activity in the future is likely to occur with the greatest frequency following glaciations, although infrequent but significant seismic events during inter-glacial periods cannot be excluded. Assuming a repetition of the last glaciation, the next glacial retreat will be in about 70,000 years time, although there are significant uncertainties, particularly in regard to the long-term impact of anthropogenic emissions, especially greenhouse gases. The probability of an earthquake occurring that is sufficiently large to cause canister damage in a 100,000 year time frame has been estimated as 0.02.

Radionuclide release and transport analyses

The majority of canisters are expected to provide complete containment of radionuclides over a prolonged period in all identified scenarios. However, since the possibility of one or more canister failures cannot currently be excluded over a million year time frame, the consequences of canister failure have to be assessed, taking into account uncertainties in the mode of failure and subsequent radionuclide release and transport processes. Thus, a range of assessment cases – i.e. specific model realisations of different possibilities or illustrations of how a system might evolve and perform in the event of canister failure – has been defined and analysed in terms of hazard to humans and to other biota. The assessment cases address each identified canister failure mode: (i), an initial penetrating defect, (ii), canister failure due to corrosion and (iii), canister failure due to rock shear. For each canister failure mode, a Base Case has been defined against which to compare the results of variant assessment cases that illustrate the impact of specific uncertainties on the radiological consequences of canister failure. Parameters in the Base Cases are, in most instances, selected to be either realistic or moderately conservative in the sense that they are expected to lead to an overestimate of radiological consequences. Perturbations to radionuclide release and transport caused, for example, by the steel and cementitious components of the KBS-3H repository external to the canisters are assumed to be negligible in the Base Cases.

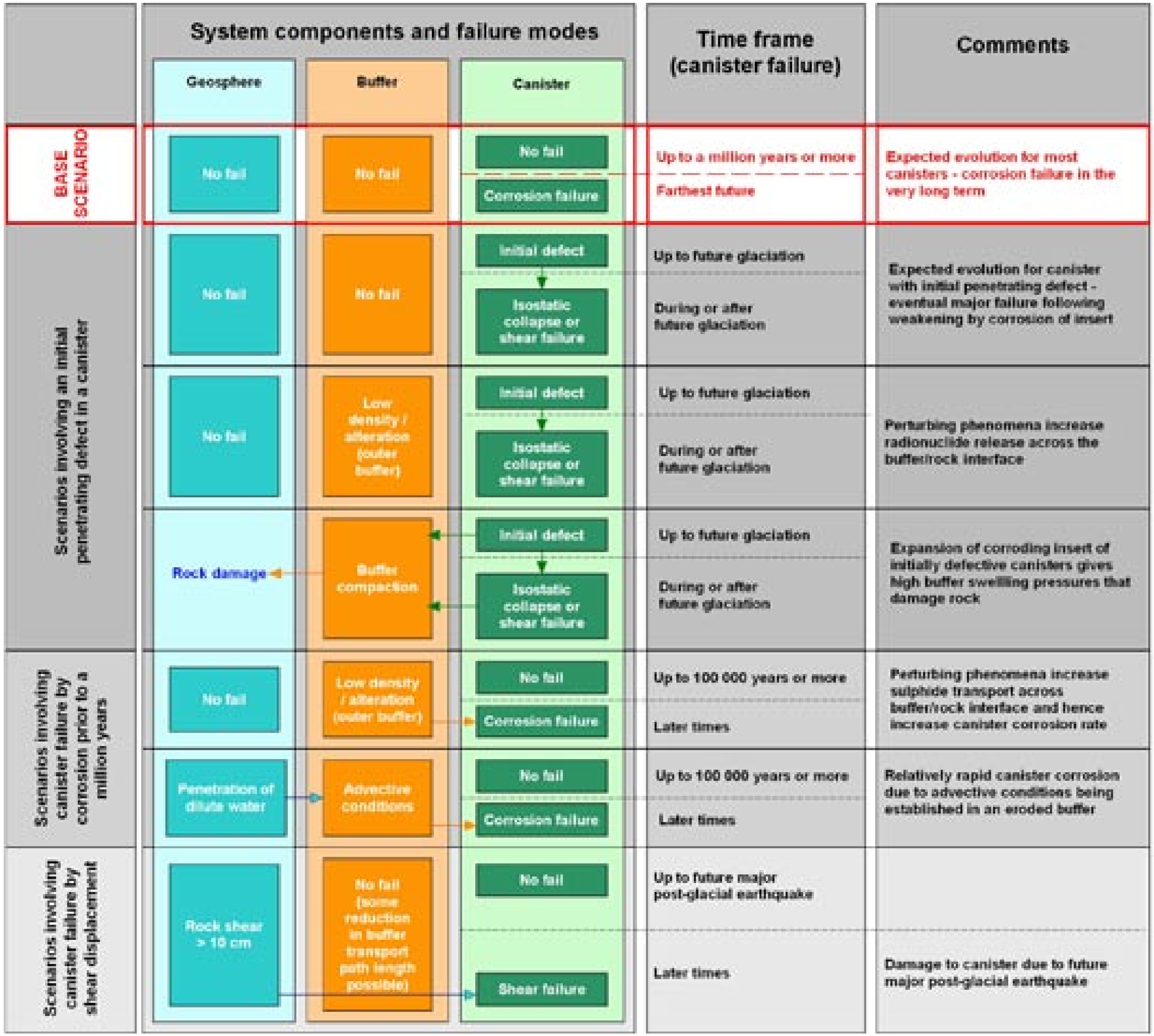


Figure 1. *Potential system states in different time frames analysed in the present safety case – the Base Scenario is shown in red.*

The variant cases are chosen to cover the various scenarios that have been identified as leading to loss or major degradation of repository safety functions, and to canister failure. In addition to these "scenario uncertainties", there are additional uncertainties that have a more limited impact on the repository safety functions. Based on the KBS-3V/KBS-3H difference analysis approach, a limited number of assessment cases are defined addressing uncertainties related to features and processes that are specific to KBS-3H, or are significantly different in KBS-3H and KBS-3V. Additional cases are also analysed to illustrate the impact of other uncertainties in key features of the safety concept.

The variant cases for the most part take a more pessimistic view of uncertainties than the Base Cases. Various measures, including the use of process tables as check lists, have been used to ensure that no important processes and associated uncertainties have been overlooked in the identification of scenarios and assessment cases.

In evaluating the assessment cases, extensive use has been made of SR-Can parameter values and model assumptions, except where these are affected by differences in the materials to be disposed of in Finnish and Swedish repositories, and differences between conditions at Olkiluoto and those at the Swedish sites considered in SR-Can. Where differences arise, the selection of parameter values and model assumptions has been made largely according to "expert judgement", based on considerations such as use in previous assessments, additional data gathering and laboratory studies. In the case of geosphere transport modelling, the modelling approach and parameter values used are based largely on TILA-99, although more recent developments in the understanding of the Olkiluoto site are used to provide additional support for the parameter values selected (for example, in terms of their conservatism).

The primary assessment endpoints in the present safety assessment are:

- annual effective dose[2] to most exposed individual considering multiple exposure pathways in the biosphere, which is used for comparison with the Finnish regulatory dose criterion for the "environmentally predictable future",
- activity fluxes to the biosphere (geo-bio fluxes) which are used for comparison with Finnish regulatory geo-bio flux constraints.

In addition, a safety indicator based on an indicative stylised well scenario that considers only the drinking water pathway – WELL-2007 dose – has been calculated for all assessment cases. Calculation of WELL-2007 dose further facilitates comparison with regulatory guidelines for the "environmentally predictable future", as well as the results from other safety assessments and safety cases, without the need to justify a wide range of biosphere modelling assumptions.

In all cases, the calculational results comply with Finnish regulatory criteria.

A comparison of the results of the KBS-3H safety assessment calculations with those of TILA-99 and SR-Can confirms that they are consistently in the same range. Moreover, where there are differences between results, it is possible to identify the reasons.

Complementary evaluations of safety

Complementary lines of reasoning have been developed that lend support to the long-term safety of the KBS-3H repository at Olkiluoto. These include:

(i) studies of natural and archaeological analogues, which support the evaluation of total system performance as well as understanding and confirmation of important, safety-relevant processes,

(ii) evidence for the geological stability of the Olkiluoto site,

[2] In the present safety assessment, the annual effective dose to the most exposed individual considering multiple exposure pathways is termed the annual landscape dose.

(iii) analyses of the decreasing radiological toxicity[3] of the spent fuel over time, which can be placed in perspective by comparing it with the radiological toxicity of naturally occurring radioactive materials, such as uranium ore bodies,

(iv) comparisons of calculated doses due to the repository with doses arising from natural background radiation and consumption of ordinary drinking water, which indicate that they are insignificant (and only moderate even in the hypothetical case of the simultaneous failure of all canisters),

(v) comparisons of calculated radiotoxicity fluxes[4] from the repository with those of natural radionuclides moving in groundwater and other pertinent radiotoxicity fluxes, confirming the insignificance of the calculated repository releases,

(vi) consideration of the hazards due to the chemotoxicity of releases from the repository, suggesting that these are also insignificant.

Conclusions

Overall, the conclusions of the long-term safety assessment of a preliminary design for a KBS-3H repository for spent nuclear fuel at Olkiluoto are as follows.

1. In the absence of any initial penetrating defect in the canisters, no canister failures should occur during the first several thousand years after canister deposition provided the repository system evolves as expected. Thereafter, the processes that are potentially the most detrimental to repository safety are related to glacial conditions. This was also a main conclusion arising from SR-Can in the case of a KBS-3V repository for spent fuel at two Swedish sites, but the importance of some geosphere properties may differ, e.g. the KBS-3H design is more sensitive to sub-vertical fractures with respect to potential damage to the engineered barrier system by rock shear.
2. Safety issues related to a future change to glacial conditions at the Olkiluoto site are generally the same as those identified in SR-Can for the KBS-3V design at Swedish sites, the most significant being canister failure due to rock shear in the event of a large, post-glacial earthquake and loss of buffer from exposure to glacial meltwater, which may lead to early failure of some canisters by corrosion. There are, however, some differences compared with SR-Can and KBS-3V, e.g. the probability of, and possibility of avoiding by design, fractures that can undergo rock shear movements that damage canisters in the event of a large post-glacial earthquake. Furthermore, in the case of KBS-3H, loss of buffer around one canister due to exposure to glacial meltwater may affect the corrosion rate of neighbouring canisters, since the the buffer density along the drift will tend to homogenise over time. This also means that the impact on buffer density and on the corrosion rate of the first canister will diminish with time. In the case of KBS-3V, on the other hand, buffer loss around one canister will not affect the state of the buffer around the other canisters.
3. A difference analysis has shown that the key differences in the evolution and performance of the KBS-3H and KBS-3V designs relate mainly to the engineered barrier system and to the impact of local variations in the rate of groundwater inflow on buffer saturation along the drifts. The safety functions of the geosphere are generally not expected to differ significantly between the two designs.
4. No features or processes that are specific to KBS-3H have been identified that could lead to a loss or substantial degradation of the safety functions of the engineered barriers over a million year time frame. However, the degree to which fractures with the potential to undergo

[3] Radiotoxicity is a measure of the potential toxicity due to ionising radiation (often referred to as the "hypothetical dose") following ingestion or inhalation of a radionuclide. It is often used in the form of radiotoxicity index where the total hypothetical dose for all radionuclides is compared to a regulatory guideline or criterion.

[4] The radiotoxicity flux is a measure of the hazard created by a flux of radionuclides across a defined interface within a given period (see Chapter 10). In the form used in this report, it is effectively the radiotoxicity index per square kilometre per year.

shear movements that damage the engineered barriers in the event of a large earthquake can be identified and avoided remains to be evaluated, and may be different for KBS-3H compared with KBS-3V.

5. Particularly in tight drift sections, the gas generated by the steel components of the KBS-3H repository external to the canister in the current reference design (principally the supercontainer shell) may accumulate at the buffer/rock interface, possibly resulting in a prolonged period during which significant inflow of water from the surrounding rock will be limited and the buffer will remain only partially saturated.
6. The timing of eventual canister failure by corrosion may be affected by perturbations to the buffer/rock interface caused, for example, by the presence of the steel supercontainer shell and its corrosion products. The issues related to the impact of iron and its corrosion products on the buffer bentonite are potentially detrimental to the safety functions of the buffer and subject to significant uncertainties. Hydrogen generation during the first thousands of year may also affect the corrosion of the canisters via its effect on the microbial reduction of sulphate to sulphide, but the sulphides formed may be precipitated as iron sulphides by reacting with the iron corrosion products, thus reducing the flux of sulphide to the canister surface. The conclusions from the analyses performed are that these perturbations are not expected to lead to canister failure by corrosion within a million year time frame.
7. Radionuclide release from the repository near field in the event of canister failure may also be affected by perturbations to the buffer/rock interface, but in all cases releases are limited and comply with Finnish regulatory criteria. Only single canister failure cases have, however, been considered and the possibility of multiple canister failures must be addressed in future studies.
8. Several issues have been identified for further study, many of which are relevant to both KBS-3V and KBS-3H. These include, for example, site-specific issues such as the transport rate of abiogenic methane and the kinetics of sulphate reduction in the rock. While some issues, such as those related to gas generation prior to canister failure, are relevant mainly to KBS-3H, it should also be noted that there are some issues that are specific to KBS-3V.

These conclusions are based on the analysis of a KBS-3H reference design, termed the Basic Design and its application to the Olkiluoto site. It should, however, be emphasised that this choice of reference design is preliminary, and that design alternatives are being developed.

The present safety assessment has some important limitations, including that the feasibility of implementing the current reference design has been assumed, even though several design issues remain to be addressed. Nevertheless, it can be concluded, based on the present safety assessment, that the KBS-3H design alternative offers potential for the full demonstration of safety for a repository at Olkiluoto site and for the demonstration that it fulfills the same long-term safety requirements as KBS-3V. Studies are being undertaken to address remaining critical scientific and design issues. These include the further development of the DAWE design alternative to avoid the possibility of distance block displacement or deformation, which could lead to significant piping and erosion, as well as studies of iron/bentonite interaction and the possible use of alternative materials, such as titanium for the supercontainer shell and some other engineered structures in the drift.

1 Introduction

1.1 KBS-3H long-term safety studies

The KBS-3 method, based on multiple barriers, is the proposed spent fuel disposal method both in Sweden and Finland. There are two design alternatives for the KBS-3 method: KBS-3V in which the canisters are emplaced in individual vertical deposition holes and KBS-3H in which several canisters are emplaced in horizontal deposition drifts (see Figure 1-1). The reference alternative for the implementing organisations, SKB in Sweden and Posiva in Finland, is KBS-3V. Posiva and SKB have conducted a joint Research, Demonstration and Development (RD&D) programme in 2002–2007 with the overall aim of establishing whether KBS-3H represents a feasible alternative to KBS-3V.

The RD&D programme has included various studies relevant to *long-term* or *post-emplacement* safet*y*, i.e. safety from the time of emplacement of the first canisters in the repository. Construction and operation of the repository drifts will continue over several decades following emplacement of the first canisters, and long-term safety studies consider evolution and performance in this period, as well as in the period subsequent to repository closure. The safety of the workforce and the public during construction, operation and closure of the repository (operational safety) is, however, considered separately from the long-term safety studies, and will be addressed in the Design Description 2007 report by Autio et al. 2008. Throughout this report, the terms "safety studies" and "KBS-3H safety studies" refer to the KBS-3H long-term safety studies described here.

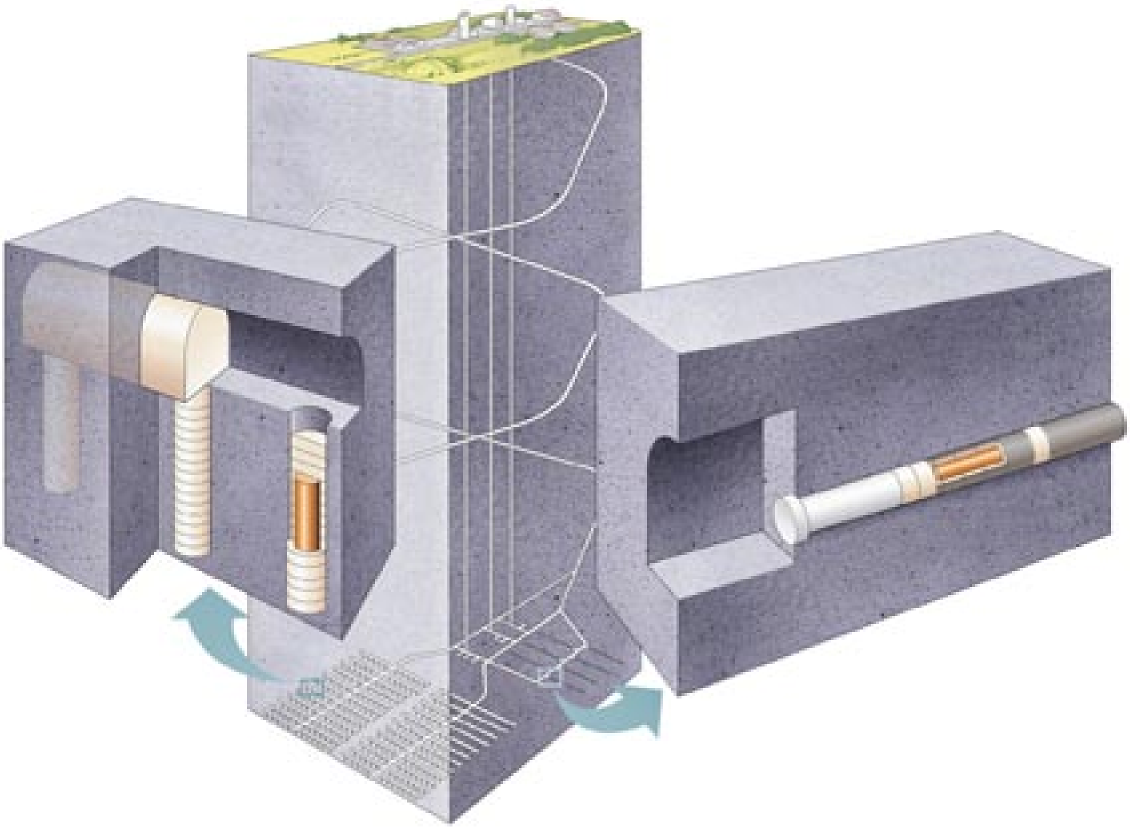


Figure 1-1. *The KBS-3V (left) and KBS-3H (right) alternatives of the KBS-3 spent fuel disposal method.*

Safety studies have been complemented by detailed studies of:

- the function of the bentonite buffer,
- repository design and layout adaptation to the Olkiluoto site in Finland,
- deposition equipment,
- the retrievability of the canister in KBS-3H,
- the comparative costs of the KBS-3H and KBS-3V alternatives.

These are intended to be sufficiently comprehensive that they can be used, along with other technical demonstration, environmental and cost studies, as a technical basis for a decision in 2008 on whether or not to continue the development of KBS-3H.

There are currently a number of design variants under consideration for implementing KBS-3H, as well as differences between the fuel types, between the characteristics and inventories of the canisters, and between the designated repository site for a spent fuel repository in Finland and the sites under consideration in Sweden. In order to focus the KBS-3H safety studies, however, they are applied to the Olkiluoto site, in the municipality of Eurajoki, which is the proposed site for a spent fuel repository in Finland. Three fuel types are considered: VVER-440 PWR fuel from the Loviisa 1 and 2 reactors, BWR fuel from the Olkiluoto 1 and 2 reactors and EPR fuel from Olkiluoto 3 (the reference fuel type for the majority of safety assessment calculations is the BWR fuel from Olkiluoto 1-2). The current basis for safety assessment is that disposal of approximately 5,500 tU fuel will be required, encapsulated in approximately 3,000 canisters. The reference KBS-3H repository design for safety assessment is the Basic Design as described in the Design Description 2006 /Autio et al. 2007/. This choice of reference design is preliminary and other design variants, also based on KBS-3H, are presented in the Design Description 2006. At the time of selection of the reference design for the long-term safety studies, no major differences between the Basic Design and the design alternative, termed DAWE (Drainage Artificial Watering and air Evacuation), had been identified that were relevant to long-term safety. The Basic Design is the outcome of several years of studies of different design options for the drift. DAWE was introduced at a later stage in the programme to address some uncertainties regarding the feasibility of implementing the Basic Design in less favourable locations along the drifts. At the time of selection, however, both designs were judged to be potentially feasible, and the Basic Design was selected for the safety assessment.

Specific high-level questions addressed by the KBS-3H safety studies are:

- Are there safety issues specific to KBS-3H with the potential to lead to unacceptable radiological consequences?
- Is KBS-3H promising at a site with the broad characteristics of Olkiluoto from the long-term safety point of view?

The safety studies are, however, limited in scope and do not currently address the questions:

- Is KBS-3H more or less favourable than KBS-3V from a long-term safety point of view?
- Does the specific realisation of the KBS-3H repository design considered in the safety studies satisfy all relevant regulatory guidelines?

Regarding the first question, a comparative study of favourable and less favourable features of KBS-3H and KBS-3V is beyond the scope of the safety studies carried out so far. Regarding the second question, although the performance of a KBS-3H repository has been analysed for a number of variants cases illustrating the impact of different uncertainties and the results compared with Finnish regulatory guidelines, the analyses are not comprehensive in their consideration of uncertainty, and leave a number of issues unresolved, as described at length in the KBS-3H Radionuclide Transport Report /Smith et al. 2007a/. These limitations would have to be addressed before it could be judged whether or not all relevant regulatory guidelines are satisfied.

The feasibility of implementing the Basic Design is assumed in the safety studies, although several design issues remain to be addressed, as discussed in Section 11.4. The feasibility of implementing the chosen design must be justified as part of any future safety case. Feasibility issues are discussed in the Design Description 2007 report /Autio et al. 2008/.

1.2 Reporting of KBS-3H safety studies

In order to judge feasibility of implementing KBS-3H from a long-term safety point of view, relevant safety issues must be understood as well for KBS-3H as they are for KBS-3V. There is a broad scientific and technical foundation that is common to both alternatives, and much of the work carried out by both Posiva and SKB in the context of KBS-3V is also directly applicable to KBS-3H (Section 1.3). Thus, there is comparatively much more limited documentation that has been developed specifically relating to KBS-3H, and this documentation focuses primarily on the differences identified between KBS-3H and KBS-3V in a systematic "difference analysis" approach.

The several reports that document and support the safety studies of a KBS-3H repository at Olkiluoto are shown in Figure 1-2 (although some are common to the KBS-3H and KBS-3V and will be developed in the context of Posiva's KBS-3V programme, see Section 1.3). The reporting structure of the KBS-3H safety studies is based on Posiva's safety case plan /Vieno and Ikonen 2005/.

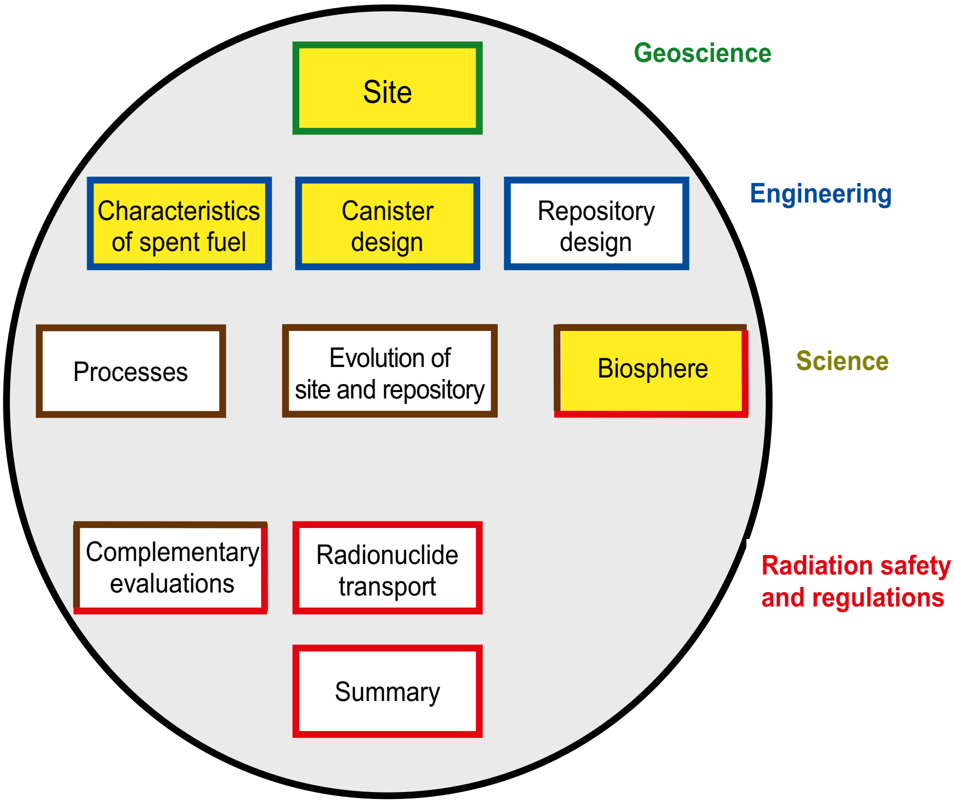


Figure 1-2. *The reporting structure for KBS-3H long-term safety studies. The colours of the boxes indicate the areas covered by the reports (as listed on the right-hand side of the figure). Yellow filling indicates reports common to the KBS-3H and -3V safety studies. All the other boxes represent reports produced within the KBS-3H safety studies or design studies. The safety assessment for a KBS-3H spent nuclear fuel repository at Olkiluoto is presented in the Summary report. For details see the main text.*

The overall outcome of the KBS-3H safety studies is documented in the present "Safety assessment for a KBS-3H spent nuclear fuel repository at Olkiluoto – Summary Report" (referred to as **summary report** in Figure 1-2). The summary report is supported by a number of further high-level reports (those shown in Figure 1-2).

The geoscientific basis of the safety case is provided in Olkiluoto **site reports** /Posiva 2003, 2005, Andersson et al. 2007/, including the present situation at, and past evolution of, the Olkiluoto site, and disturbances caused by ONKALO[5], an underground characterisation facility that will also serve as an access route to the repository. Data from the most recent Olkiluoto Site Description 2006 /Andersson et al. 2007/ are referenced in the present report, although further work is required to incorporate these data fully in future safety assessments.

The engineering basis is provided by the reports on the **characteristics of spent fuel** /Anttila 2005a/, **canister design** /Raiko 2005/, and **repository design** /Autio 2007, Autio et al. 2007/, which are further described in Section 1.3.1. The preliminary "reference design" analysed in the present safety assessment is presented in the repository design report titled "Design Description 2006" /Autio et al. 2007/. The reference design for the KBS-3H safety studies (termed Basic Design) was frozen at the beginning of 2007. The KBS-3H repository design is still ongoing and subsequent design developments are presented in the Design Description 2007 report /Autio et al. 2008/.

The scientific understanding supporting the safety studies is synthesised in a **Process Report** /Gribi et al. 2007/ and in an **Evolution Report** /Smith et al. 2007b/. The Process Report, as its name indicates, describes the individual processes and discusses the relevance of selected processes (e.g. gas generation) through scoping calculations. The processes discussed in the Process Report are selected based on the comparison with the KBS-3V processes discussed in the SR-Can safety assessment /SKB 2006a/, according to the difference analysis approach mentioned above. The Evolution Report describes the same processes, but in broadly chronological order, highlighting the interactions between the processes and their coupling whenever possible, starting from repository construction and continuing up to one million years from the beginning of repository operations (evolution in the farthest future, beyond a million years, is also described, though only briefly and in qualitative terms)[6].

The Process and Evolution Reports provide the basis for the selection of the assessment cases calculated in the **Radionuclide Transport Report** /Smith et al. 2007a/. A **Complementary Evaluations of Safety Report** /Neall et al. 2007/ provides additional arguments on the long-term safety aspects of a KBS-3H repository located at the Olkiluoto site. A **Biosphere Analysis Report** /Broed et al. 2007/ was produced in parallel to the above-mentioned reports using input from the KBS-3H radionuclide transport report and the main conclusions on the ensuing releases to the biosphere are summarised in the present safety assessment summary report.

In the following sections, unless otherwise stated, "Process Report" refers to /Gribi et al. 2007/, Evolution Report to /Smith et al. 2007b/, Radionuclide Transport Report to /Smith et al. 2007a/ and Complementary Evaluations of Safety Report to /Neall et al. 2007/.

These high-level reports are further supported by more detailed technical reports compiled in support of KBS-3H safety studies (see also Section 1.3.2), including reports on thermal analyses /Ikonen 2003, 2005/, thermo-mechanical analyses /Lönnqvist and Hökmark 2007/, layout studies, based on analyses of data from the Olkiluoto site /Hellä et al. 2006/, discrete

[5] ONKALO is the Olkiluoto Underground Rock Characterisation facility for site-specific underground investigations. ONKALO has been under construction since mid-2004 and will serve as an access route to the repository and the first disposal tunnels are planned to be adjacent to ONKALO's main characterisation level.

[6] A certain degree of overlap between the Process and the Evolution Reports is unavoidable and deemed beneficial if the reports are read separately. However, the authors recognise that it is preferable to read both the Evolution and Process Reports together to fully grasp the couplings and relative importance of processes in such a complex system.

fracture network modelling of the site /Lanyon and Marschall 2006/, analyses of HMCGB (hydro-mechanical-thermal-gaseous-microbiological) processes related to the steel components /Johnson et al. 2005/, experimental and modelling studies on the interaction of iron and bentonite /Carlson et al. 2006, Wersin et al. 2007/, and solubility estimation in support of radionuclide release and transport calculations /Grivé et al. 2007/.

A project decision was taken not to prepare a separate data report for the KBS-3H safety studies (in contrast to the case of SR-Can, see /SKB 2006b/). Instead, all main data used the reports of these safety studies, excluding input parameter values for radionuclide release and transport calculations, are reported in Appendix A of the Process Report. The data are based on the information available at the time of report writing (2006–2007), including the preliminary design information presented in the KBS-3H Design Description 2006 /Autio et al. 2007/, laboratory data, field data, modelling, calculations, and in some cases, expert judgment. The basis for data selection and assumptions used has been reported as far as possible in the Process Report. Input parameter values for radionuclide release and transport calculations, and the basis for their selection, are given in the Radionuclide Transport Report.

1.3 Scientific basis of KBS-3H safety studies

There has been around 30 years of scientific R&D and technical development in the Swedish and Finnish KBS-3V programmes, much of which is directly applicable to KBS-3H project, allowing work on KBS-3H to focus on those issues that are unique to this alternative. The scientific basis common to the two alternatives and that developed specifically for KBS-3H are summarised below.

1.3.1 Scientific basis common to KBS-3H and KBS-3V

The investigations of the Olkiluoto site over the last 20 years, and the distillation of understanding into the Site Description 2006 /Andersson et al. 2007/, have been carried out to support the development of a KBS-3V repository but are clearly available and also applicable to the KBS-3H repository safety studies. For the surface conditions, the Biosphere Description 2006 /Haapanen et al. 2007/ complements the information in the Site Description.

Other important information acquired in the context of KBS-3V that is also relevant to KBS-3H includes the following items.

a. Evolution of the Site and Repository for KBS-3V /Pastina and Hellä 2006/

This report is the first detailed description of the expected evolution of a KBS-3V repository at the Olkiluoto site. The KBS-3V Evolution Report will be one element of a future safety case for such a repository. The repository is designed for the disposal of the same quantity of spent fuel (approximately 5,500 tU), encapsulated in the same copper canisters, as the KBS-3H repository considered in the present safety assessment. The report may be considered a template for the KBS-3H Evolution Report /Smith et al. 2007b/ as many of the phenomena discussed are the same for both repositories.

b. Terrain and Ecosystems Development Model of Olkiluoto Site /Ikonen et al. 2007/ and Landscape Model of Radionuclide Transport in the Biosphere /Broed 2007/

The report by /Ikonen et al. 2007/ describes forecasts of the evolution of the surface environment over time, including shoreline displacement (which is also summarised in the KBS-3V Evolution Report). The model constructed to simulate the radionuclide transport and fate in the biosphere in the KBS-3H assessment /Broed 2007/ is based on these forecasts.

c. Characteristics of spent fuel /Anttila 2005a/

Radioactive characteristics of the three different Finnish spent fuel types (VVER-440, BWR and EPR fuel bundles) have been calculated. For each fuel type, the initial enrichment and the discharge burnup have been varied in order to determine the range of properties which the eventual spent fuel could exhibit. The main results of the calculations include the radionuclide composition of spent fuel and cladding, decay heat production and radiation (photon and neutron) source terms of the spent fuel as a function of cooling time (at 51 time points).

d. Canister design /Raiko 2005/

The report provides a summary of the design of the canister for final disposal of Finnish spent nuclear fuel. The canister structure consists of a cylindrical massive nodular graphite cast iron insert covered by a 50 mm thick copper overlay. The canister has three versions, one for each reactor type in Finland. The fuel is sealed into the canisters in whole fuel assemblies including the possible flow channel outside the bundle. The capacity of the canister is 12 assemblies of BWR or VVER-440 fuel and 4 assemblies of EPR fuel. The report describes the development ofthe design basis for the canister, summarises the analyses performed and discusses the results.

e. KBS-3V Repository Design /Saanio et al. 2004, 2006/

These reports describe the preliminary designs (Stages 1 and 2, respectively) for a KBS-3V repository for spent fuel at a depth of 400–500 metres at Olkiluoto. Much of the infrastructure could be identical with that for a KBS-3H repository, including access routes to the repository (an inclined access tunnel and vertical shafts), the encapsulation plant above ground, central tunnels and technical rooms at the disposal level. In addition, much of the support services, such as power, ventilation and drainage, will be similar despite differences in the repository design. Although there will be differences in detail, for example, for controlled and uncontrolled area planning, between the two repositories, these reports set out much of the information that is also needed for the KBS-3H repository designs.

f. Reports prepared as part of the Swedish SR 97 Safety Assessment /Andersson 1999, Lindgren and Lindström 1999, SKB 1999abc/

In preparation for site investigations for siting of a deep repository for spent nuclear fuel, the Swedish Government and nuclear regulatory authorities requested an assessment of the repository's long-term safety to demonstrate that the KBS-3 method was likely to meet the safety and radiation protection requirements. The SR 97 safety assessment was the outcome. The purpose was to demonstrate by means of a systematically conducted analysis that the risk of harmful effects to individuals in the vicinity of the repository complies with the acceptance criterion formulated by the Swedish regulatory authorities, i.e. that the risk may not exceed 10^{-6} per year. Geological data were taken from three sites in Sweden to demonstrate a range of different conditions in Swedish granitic bedrock. The repository is of the KBS-3V type constructed at a depth of 500 m in the bedrock. A considerable number of supporting reports brought together state-of-the-art information to support the safety assessment. A key report was the SR 97 Process Report /SKB 1999c/.

g. Posiva's localisation of SR 97 Process Report to Olkiluoto /Rasilainen. 2004/

The SR 97 Process Report described and assessed all the thermal, hydrological, mechanical, chemical and biological processes potentially occurring in a KBS-3V repository in a systematic way according to the locus of the process in the repository system. This methodical analysis used data from three different Swedish sites as well as state-of-the-art understanding of the processes themselves. The similarity of the KBS-3V repository designed for the Olkiluoto site provided an opportunity to consider the same processes while concentrating on data from the Finnish example. This resulted in a process report that is a localisation of the SR 97 Process

Report /SKB 1999c/ to Posiva's current situation at Olkiluoto and contains a commentary which concentrates explicitly on Finnish observations.

h. SR-Can and supporting reports /SKB 2006abc/

The Swedish SR-Can project was a preparatory stage for the SR-Site assessment that will be used in support of SKB's application for a final repository. Two purposes of the SR-Can safety assessment were to make a first assessment of the safety of potential KBS-3V repositories at Forsmark and Laxemar to dispose of spent fuel in copper canisters and to provide feedback to design development, to SKB's R&D programme, to further site investigations and to future safety assessment projects.

As was the case with the preceding SR 97 study, the SR-Can safety assessment was supported by a large number of auxiliary reports in which state-of-the-art information and system understanding were brought together. Moreover, there was also much effort made to demonstrate that the methodology of the assessment was both adequate to the task and conformed to the requirements of the relevant safety regulations.

The SR-Can supporting reports have provided important input the KBS-3H safety studies and, to an extent, allowed these studies to concentrate on assessing the impact of the differences of KBS-3H compared with KBS-3V and on the influence of the specific properties of the Olkiluoto site.

i. International FEP Database /NEA 2000/

This internationally reviewed and revised compilation of FEPs /NEA 2000/, which includes FEP databases from several national programmes and international exercises, provided a checklist against which the FEPs identified in the KBS-3H Process Report can be compared to ensure completeness at the outset.

1.3.2 Investigations specific to KBS-3H

The preceding section described the most substantial, relevant sources of information available to the KBS-3H safety studies from the long-established KBS-3V programme in both Finland and Sweden. However, the differences between the alternatives have naturally required new studies to be carried out to address KBS-3H specific processes and issues. The following studies include those already completed and published as well as others more recently initiated that are currently ongoing.

a. HMCBG Processes related to the Steel Components in the KBS-3H Disposal Concept /Johnson et al. 2005/

An analysis of the hydro-mechanical-chemical-biological processes affected by gas (HMCBG) related to the steel components in the KBS-3H deposition drift has been performed. The outcome of this study contributes to the KBS-3H Process Report for a repository for spent fuel sited at Olkiluoto. The steel components, i.e. the supercontainer, distance block fixing rings and compartment seals, are specific to KBS-3H and have no equivalent in KBS-3V. Consequently, an assessment of processes arising from the behaviour of these components is unique to KBS-3H studies.

b. Thermal analyses for a KBS-3H type repository /Ikonen 2003, 2005/

Thermal analyses for a KBS-3H type repository at Olkiluoto have been performed by /Ikonen 2003, 2005/. Particular attention has been paid to the effects of gaps between the canister and buffer and between the buffer and the supercontainer shell and rock, and to the thermal dimensioning of the repository.

c. Thermo-mechanical analyses of a KBS-3H deposition drift at the Olkiluoto site /Lönnqvist and Hökmark 2007/

Thermo-mechanical (TM) modelling was performed by /Lönnqvist and Hökmark 2007/ to study the risk of rock spalling and the degree of swelling buffer pressure on the deposition drift wall that may cause opening of intersecting rock fractures in the case of a KBS-3H deposition drift at Olkiluoto. Results indicate that, in a reference layout in which the deposition drifts are aligned with the direction of the principal stress in the rock, little or no spalling will occur prior to the emplacement of spent fuel, but that thermally-induced rock spalling after a few years of heating is possible if there is no support pressure on the drift wall due, for example, to buffer swelling. They also indicate that a pressure on the drift wall of at least 10 MPa is required to open a pre-existing horizontal fracture intersecting the drift at mid-height, although the effects are expected to be small in terms of increase in fracture aperture and distance from the drift wall to which such effects extend, even at pressures of 20 to 25 MPa.

d. Studies of Buffer Behaviour in KBS-3H Concept /Börgesson et al. 2005/

In KBS-3H the bentonite buffer is emplaced partly within the supercontainers and partly as the intervening distance blocks (Section 4.1). Both these components are substantially different from the buffer as emplaced in KBS-3V. In particular, the expectation that the bentonite within the supercontainer will extrude to fill the open space to the drift walls and the requirement for the distance blocks to seal and isolate the individual supercontainers have no exact equivalent in KBS-3V. As a result, the implications for the final saturated state of the buffer of, for example, localised water inflow into the deposition drift must be assessed specifically for KBS-3H. The function of the buffer in KBS-3H was investigated by laboratory tests at small and full scale, by modelling and by scenario analyses and the results are reported by /Börgesson et al. 2005/. The new summary report by /Sandén et al. 2008/ summarises more recent buffer test results from the period 2005–2007.

e. DFN (Discrete Fracture Network) Modelling, dealing with Groundwater Flow through the Fracture Network at Olkiluoto for a KBS-3H type repository /Lanyon and Marschall, 2006/

This report presents Discrete Fracture Network (DFN) models of groundwater flow around a KBS-3H repository situated at Olkiluoto. The study was performed in support of the KBS-3H safety studies and, in particular, of the Process and Evolution Reports. In the course of the task definition, the need was identified for complementary modelling studies aimed at increasing insight into the hydrodynamic evolution of the disposal system during and after the operational period. Input from hydrodynamic models was required in order to assess the probability of high initial groundwater inflow points, which could, if not avoided by design, cause buffer erosion, and to assess the evolution of inflows after construction of deposition drifts, including the effects of inflow to a drift of neighbouring open drifts and transport tunnels.

f. Experimental Studies of the Interaction of Anaerobically Corroding Iron and Bentonite /Carlson et al. 2006/

The KBS-3H repository has a number of steel components that have no equivalent in the KBS-3V repository. These components are largely in direct contact with the bentonite of the buffer and, particularly in the case of the supercontainer, present a very large surface area for potential interaction as the steel corrodes. This is a different situation from KBS-3V where the iron insert within the copper canister is expected to have minimal contact with bentonite after canister failure. Thus, new experimental and modelling studies have been necessary to determine initially the type and extent of interaction between the steel and bentonite. These experiments have determined the mineralogical changes which could take place in the bentonite but the rate of alteration and the extent to which it will alter the buffer are still uncertain.

g. Impact of Corrosion Derived Iron on the Bentonite Buffer within the KBS-3H Disposal Concept- the Olkiluoto Site as a Case Study /Wersin et al. 2007/

As noted above, the KBS-3H repository has a number of steel components that have no equivalent in the KBS-3V repository but which could potentially interact with the bentonite. While the experimental study noted above has helped to determine the mineralogical consequences, the wider issue for the KBS-3H safety studies of consequences for long-term buffer performance are addressed by /Wersin et al. 2007/. The preliminary assessment for the Olkiluoto case indicates that the iron/bentonite interaction will remain spatially restricted for very long times, because of the slow rate of diffusive transport in the buffer and the strong affinity of the clay for Fe(II) released from the corroding supercontainer.

1.4 Structure of the present report

The remaining chapters of this report are structured as follows:

Chapter 2 gives an overview of relevant regulations applicable to a spent fuel repository in Finland.

Chapter 3 identifies the different elements of the present safety assessment, which are presented in summary form in Chapters 4 to 10 and described in detail in the supporting high-level reports shown in Figure 1-2.

Chapter 4 presents the current reference KBS-3H design, the safety concept and the safety functions of the repository, and introduces the concepts of safety function indicators and safety function indicator criteria, which are important tools used in carrying out the safety assessment.

Chapter 5 describes the Olkiluoto site, the impact of repository excavation on the site, and the characteristics (initial condition) of the engineered barriers of the repository at the time that they are emplaced underground.

Chapter 6 gives an overview of the evolution of the repository in successive time frames, beginning with early evolution in a transient phase of relatively high mass and energy fluxes, followed by a period in which the repository is in a quasi-steady state, and then eventually by a period during which major climate changes, and in particular future glacial episodes, impact on repository evolution over a time frame of about a million years. The chapter also describes briefly the further evolution of the repository up to hundreds of millions of years in the future, when geological uplift and erosion may eventually expose the repository horizon to the surface.

Chapter 7 identifies key safety issues for the repository, some of which have a different significance to, or potential impact on, KBS-3H compared with KBS-3V (as realised in the current reference design). The relevance of each issue to canister integrity and to radionuclide release and transport in the event of canister failure is discussed.

Chapter 8 describes the methodology for identifying repository evolution scenarios, and the scenarios leading to canister failure and radionuclide release from the repository which arise from applying the methodology in the present safety assessment.

Chapter 9 describes the analyses carried out to quantify radionuclide release and transport in these different scenarios, taking into account a range of uncertainties, and compares the results of the analyses with Finnish regulatory guidelines.

Chapter 10 describes complementary evaluations of safety, which include argumentation that lends further support to the findings of the analyses described in Chapter 9.

Chapter 11 describes recent advances in repository design, ongoing work in both design and system understanding, and remaining issues to be resolved in future engineering and safety studies of the KBS-3H repository.

Finally, Chapter 12 presents the conclusions of this safety assessment.

A glossary is presented in the Appendix.

2 Regulatory context

2.1 Background

The present safety assessment addresses the long-term safety of a KBS-3H repository for Finnish spent fuel located at the Olkiluoto site in Finland. It is therefore appropriate to base the assessment on Finnish regulatory requirements[7].

The regulatory requirements for a Finnish spent fuel repository at Olkiluoto are set out in the Government Decision on the safety of the disposal of spent nuclear fuel /STUK 1999/ and, in more detail, in Guide YVL 8.4 issued by the Finnish regulator /STUK 2001/. These requirements are, however, currently under revision. A detailed discussion of regulatory requirements related to the safety case, including dose and geo-bio flux constraints in different time frames, is given in Posiva's TKS-2006 report on its programme for research, development and technical design /Posiva 2006/. Some key points relevant to the present safety assessment are summarised below.

2.2 Protection criteria in different time frames

Guide YVL 8.4 distinguishes between the "environmentally predictable future" (also referred to by the regulator as the next "several thousand years"), during which conservative estimates of dose must be made (i.e. estimates that tend to over-estimate dose where there is uncertainty), and the era of "large-scale climate changes" when periods of permafrost and glaciations are expected, and radiation protection criteria are based on geo-bio flux constraints, i.e. constraints on nuclide-specific activity fluxes from the geosphere.

Posiva's interpretation of the duration of the environmentally predictable future is typically 10,000 years. The annual effective dose constraint for the most exposed members of the public applicable over this time frame is 10^{-4} Sv per year, while the average annual effective doses to other members of the public should, according to the regulations, remain insignificantly low. There are also regulatory requirements on the protection of plants and animals. According to the YVL 8.4 guideline /STUK 2001/, *"exposures shall remain clearly below the levels which, on the basis of the best available scientific knowledge, would cause decline in biodiversity or other significant detriment to any living population*" and "*moreover, rare animals and plants as well as domestic animals shall not be exposed detrimentally as individuals*." Compliance with these requirements is not discussed in the present report, but is considered in the Biosphere Analysis Report /Broed et al. 2007/.

YVL 8.4 also gives a qualitative requirement that:

"*The barriers shall effectively hinder the release of disposed radioactive substances into the host rock for several thousands of years.*"

In the era of large-scale climate changes, when comparing calculated activity releases with the constraints, the calculated values can, according to Guide YVL 8.4, be averaged over 1,000 years at most. The sum of the ratios of nuclide-specific activity releases to their respective constraints must be less than one in order to satisfy regulatory requirements.

[7] The differences between the Swedish and Finnish regulatory systems are discussed in Appendix C of the Complementary Evaluations of Safety Report /Neall et al. 2007/.

Table 2-1. Geo-bio flux constraints, as set out in Guide YVL 8.4 issued by the Finnish regulator.

Radionuclides	Geo-bio flux constraints [GBq a^{-1}]
Long-lived alpha-emitting Ra, Th, Pa, Pu, Am and Cm isotopes	0.03
Se–79; I–129; Np–237	0.1
C–14; Cl–36; Cs–135; long-lived uranium isotopes	0.3
Nb–94; Sn–126	1
Tc–99; (*Mo–93 – see main text*)	3
Zr–93	10
Ni–59	30
Pd–107; Sm–151	100

Geo-bio flux constraints, as set out in Guide YVL 8.4, are shown in Table 2-1. Guide YVL 8.4 covers all the safety relevant radionuclides considered in the present safety assessment, with the exception of Mo-93. For the purposes of the present safety assessment, a geo-bio flux constraint of 3 GBq per year is assigned to this radionuclide. This is based on the activity of Mo-93 needed to give an annual dose of 0.1 mSv in the indicative stylised drinking water well scenario described in the Radionuclide Transport Report /Smith et al. 2007a/.

In the very long term, after at least several hundred thousand years, Guide YVL 8.4 states that no rigorous quantitative safety assessment is required, but the judgement of safety can be based on more qualitative considerations. The types of considerations relevant to safety in the very long term are discussed further in /Ruokola 2002/. In the present safety assessment, safety in the very long term is addressed mainly in the Complementary Evaluations of Safety Report.

2.3 Scenarios and the impact of unlikely disruptive events

According to the IAEA definition, a scenario is a postulated or assumed sequence of states defined by the safety functions that are provided by the system components /IAEA 2003/. Guide YVL 8.4 gives some indication as to the types of evolution scenarios to be considered when evaluating doses and geo-bio fluxes. It states that:

"*A scenario analysis shall cover both the expected evolutions of the disposal system and unlikely disruptive events affecting long-term safety. The scenarios shall be composed systematically from features, events and processes, which are potentially significant to long-term safety and may arise from:*

- *mechanical, thermal, hydrological and chemical processes and interactions occurring inside the disposal system;*
- *external events and processes, such as climate changes, geological processes and human actions.*"

The Guide goes on to state:

"*The base scenario shall assume the performance targets defined for each barrier, taking account of the incidental deviations from the target values. The influence of the declined overall performance of a single barrier or, in case of coupling between barriers, the combined effect of the declined performance of more than one barrier, shall be analysed by means of variant scenarios. Disturbance scenarios shall be defined for the analysis of unlikely disruptive events affecting long-term safety*".

The importance to long-term safety of unlikely disruptive events shall, according to the Guide, be assessed. According to STUK, these events are to include at least:

- boring a deep water well at the disposal site,
- core drilling hitting a spent fuel canister,
- a substantial rock movement occurring in the environs of the repository.

Section 2.4 of Guide YVL 8.4 states that, whenever practicable, estimates of the probabilities of activity releases and radiation doses arising from unlikely disruptive events impairing long-term safety should be made. These probabilities should be multiplied by the calculated annual radiation dose or activity in order to evaluate the importance to safety of an event. In order to satisfy regulatory requirements, the expectation value should remain below the radiation dose or activity release constraints referred to in Section 2.2. If, however, the resulting individual dose implies deterministic radiation impacts (dose above 0.5 Sv), the order of magnitude estimate for its annual probability of occurrence should be 10^{-6} at the most.

In the present safety studies, the likelihood and consequence of the first two of the above-mentioned events is judged not to differ significantly between KBS-3V and KBS-3H repositories (although there will be some small difference in the probability of a vertical borehole intersecting vertically compared with horizontally emplaced canisters) and these are not discussed in the present safety assessment. The impact of substantial rock movement occurring in the environs of the repository is, however, discussed in the context of large post-glacial earthquakes (Sections 7.2.5, 8.6 and 9.6.3).

3 Elements of the safety assessment

3.1 Description of the system and of its safety functions, indicators and criteria

The first step in the safety assessment is a description of the disposal system under consideration, such as a description of the current reference design for a KBS-3H repository at Olkiluoto, together with the concept of how safety is provided (the safety concept) and the safety functions that the main system components perform. From this description, in order to assess the performance and safety of a repository, it is necessary to determine the conditions under which the identified safety functions will operate as intended, and the conditions under which they will fail, or operate with reduced effectiveness. Following the methodology adopted in the Swedish SR-Can safety assessment, the present safety assessment is based on the concepts of safety function indicators and associated criteria. One or more safety function indicators are assigned to each of the safety functions of the system components. If the safety function indicators fulfil certain criteria, then the safety functions can be assumed to be provided. If, however, plausible situations can be identified where the criteria for one or more safety function indicators are not fulfilled, then the consequences of loss or degraded performance of the corresponding safety function must be evaluated as part of the safety assessment.

Although developed for a Swedish safety assessment, SR-Can, based on Swedish regulatory requirements which specify a risk criterion, the methodology is expected to be equally applicable to an assessment in which the calculational end-points are geo-bio fluxes or doses, as required by Finnish regulations. In particular, the concepts of safety functions, safety function indicators and associated criteria are applicable irrespective of the calculational end-point.

The current reference design, its safety functions, safety function indicators and associated criteria are described in the Process Report and Evolution Report and, in summary form, in Chapter 4 of the present report.

3.2 Description of initial conditions

A description of the initial conditions within and around a KBS-3H deposition drift is given as part of the Evolution Report /Smith et al. 2007b/, and is also presented, in summary form, in Chapter 5 of the present report. The description is based largely on the design specifications of the repository, on the Olkiluoto site reports, and on the layout report /Johansson et al. 2007/.

3.3 Description of processes and evolution

In order to judge the feasibility of implementing the KBS-3H repository from a long-term safety point of view, the evolution of the repository and the processes that affect it must be understood as well for KBS-3H as they are for KBS-3V. The scientific knowledge of processes relevant to the evolution and performance of a KBS-3H repository at Olkiluoto is documented in the Process Report /Gribi et al. 2007/. The Process Report is organised in the same manner as the SR-Can Process Reports and the first Posiva KBS-3V Process report /Rasilainen 2004/. Separate chapters are dedicated to each of the key repository components, namely spent fuel, canister, buffer and distance block, supercontainer and other steel structural materials, drift end plug and backfill and geosphere (see the system description in Chapter 4). The description of each component includes each of the aspects illustrated in Figure 3-1 for the example of the canister.

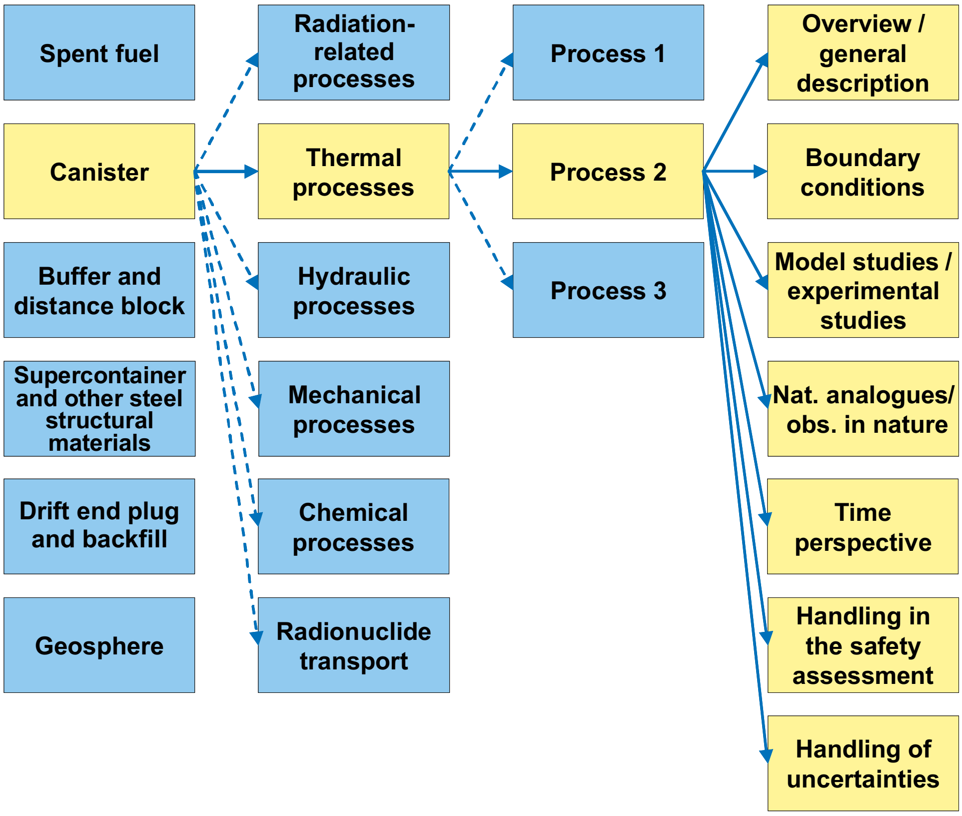


Figure 3-1. *Organisation of the process descriptions in the Process Report (example of the canister subsystem).*

Based on the scientific knowledge documented in the Process Report, the Evolution Report provides a description of repository evolution, including a description of the main uncertainties affecting this evolution. This description is also presented in summary form in Chapter 6 of the present report.

Considerable weight is put on demonstrating that the features, events and processes (FEPs) described, analysed and discussed in the Process Report allow a description of the system and its evolution to be developed to the stage that is sufficient for the purposes of safety assessment, even if the data to parameterise all processes, for example, are not available. The sufficiency of the range of FEPs considered in the Process Report has been checked by auditing the Process Report against the SR-Can Process Reports and FEP database, and also against the international FEP database maintained by the OECD/NEA, which includes FEP databases from several national programmes and international exercises /NEA 2000/. Development of Posiva's own separate FEP database for KBS-3H, in addition to the Process Report, was not considered necessary.

Although the Process Report and Evolution Report aim to be comprehensive in their coverage of safety-relevant processes, the level of detail of process descriptions varies between processes. Safety studies and detailed process descriptions for KBS-3H mainly concern those processes or issues that have a different significance to, or potential impact on, KBS-3H compared with KBS-3V. Summary descriptions are given in the Process Report and Evolution Report of processes and issues common to the two alternatives, and appropriate references are given to other reports where detailed descriptions are provided.

This approach is possible because of the broad scientific and technical foundation that is common to both alternatives (Section 1.3.1). Processes or issues that have a different significance to, or potential impact on, KBS-3H compared with KBS-3V have been identified by means of a difference analysis of safety-relevant features and processes in the two alternatives, which is also described in the Process Report.

The outcome of this analysis is summarised in Table 3-1. It shows that most of the differences between KBS-3H and KBS-3V relate to internal processes in repository deposition drifts and their immediate environment (near-field rock, drift end plugs), many of which affect how the system evolves during the early, transient phase to a state that has the desired properties after saturation. It should be noted that the difference analysis was carried out at a relatively early stage of the KBS-3H safety studies. These studies went on to identify other issues that, while not unique to KBS-3H, require additional analysis due, for example, to the fact that canister deposition in KBS-3H is in long, roughly horizontal drifts, rather than in relatively short deposition holes. Thus, the identified issues include those that could, if sufficiently extensive, in principle lead to flow and transport along the drifts, such as thermally-induced rock spalling and cement/bentonite interaction. It is these processes together with issues identified in the difference analysis that are described in most detail in the KBS-3H Process Report and Evolution Report.

3.4 Identification of key safety issues and evolution scenarios

Key safety issues identified from the descriptions of processes and system evolution given in the Process and Evolution Reports and from a review of other relevant safety assessments, particularly SR-Can, are presented and discussed in Chapter 7 of the present report. Some issues have aspects specific to KBS-3H (e.g. those related to KBS-3H-specific repository components), whereas others are common to KBS-3V and KBS-3H. All have potential implications for canister longevity or for the capacity of other system components to retain radionuclides or retard and dilute radionuclide releases in the event of canister failure.

By considering the potential impact of the key safety issues on the safety functions of the repository, various possible evolution scenarios are identified whereby one or more canister failures leads to radionuclide release and transport, and exposure of humans and other biota to released radionuclides. The scenarios and the methodology for their identification are described in the Evolution Report, and summarised in Chapter 8 of the present report.

3.5 Carrying out of radionuclide release and transport analyses

The consequences of scenarios leading to canister failure, taking into account uncertainties in release and transport processes, are assessed by defining a range of assessment cases – i.e. specific model realisations of different possibilities or illustrations of how a system might evolve and perform in the event of canister failure – and analysing these cases in terms of hazard to humans and to other biota, quantified in terms of annual effective dose[8] or geo-bio fluxes (in order to compare with Finnish regulatory guidelines). Given that a key question addressed by the KBS-3H safety studies is whether or not there are safety issues identified in the KBS-3V/KBS-3H difference analysis with the potential to lead to unacceptable radiological consequences, a number of specific assessment cases are defined addressing uncertainties related to features and processes that are specific to KBS-3H, or are significantly different in KBS-3H and KBS-3V. Additional cases are also analysed to illustrate the impact of other uncertainties in key features of the safety concept. Radionuclide release and transport processes and analyses are described in detail in the Radionuclide Transport Report /Smith et al. 2007a/ and are summarised in Chapter 9 of the present report.

[8] Exposure of other biota is not explicitly addressed in the present safety assessment, but is considered in the Biosphere Analysis Report /Broed et al. 2007/.

Table 3-1. Major differences identified in the difference analysis of safety-relevant features and processes in KBS-3V and KBS-3H. The difference analysis was carried out at a relatively early stage of the KBS-3H safety studies. These studies went on to identify a number of further issues (see main text).

System components/ (groups of) processes	KBS-3V	KBS-3H
Copper canister, cast iron insert, fuel/cavity in canister		
The canister, insert and fuel are the same in both alternatives		
Buffer		
Piping/erosion by water and gas, chemical erosion	Within deposition hole at buffer/rock interface in the case of high initial inflow rates (however, the holes can be selected individually and those with larger inflows will be rejected). Also, in the longer term, chemical erosion is possible in the event of an influx of glacial meltwater. Loss of buffer around one canister due to piping/erosion or chemical erosion by glacial meltwater will not affect the buffer around neighbouring canisters.	Piping/erosion may affect buffer density at bentonite/rock interface in canister sections with high initial inflow rates and in canister sections adjacent to these; mitigating the effects of piping/erosion is considered to be a major challenge in the design of KBS-3H and has led to the consideration of two candidate designs and various design alternatives. Deposition drift sections with inflows larger than a specified limit are not used for deposition – but sealed tightly. This will affect the utilisation degree of deposition drifts. Design is still under development /Autio et al. 2007/. Chemical erosion is possible in the event of an influx of glacial meltwater. Loss of buffer around one canister due to piping/erosion or chemical erosion by glacial meltwater may affect the buffer around neighbouring canisters, since the the buffer density along the drift will tend to homogenise over time.
Displacement of buffer/distance block (leading to a reduction in bentonite density)	Swelling of buffer from deposition hole into drift above the hole may lead to lowering of bentonite density; rock stress distribution leads to risk of rock slabs at mouth of deposition hole.	Axial displacement of distance block by hydraulic pressure build-up may lead to lowering of bentonite density and must be counteracted by a rapid emplacement rate and by the use of steel plugs and steel rings bolted to rock, as described in the current reference design /Autio et al. 2007, Börgesson et al. 2005/. Axial displacement due to heterogeneous swelling is limited by friction and by drift end plug.
Iron/bentonite interaction	Relevant only for failed canisters.	In addition to the processes relevant to KBS-3V, significant geochemical interactions between supercontainer and buffer will take place (iron/smectite interaction, iron-silicate formation, cation exchange, etc); these processes may affect the buffer density, swelling pressure, hydraulic conductivity and other properties; these effects are locally limited at early times, but may develop with time and affect larger parts of the buffer /Johnson et al. 2005, Carlson et al. 2006, Wersin et al. 2007/.
Gas transport and possibly gas-induced porewater displacement	Relevant only for failed canisters.	In addition to the processes relevant to KBS-3V, significant gas effects are expected /Johnson et al. 2005/ due to anaerobic corrosion of supercontainer and other steel components (retarded resaturation, air trapping, gas dissolution/diffusion/advection, gas pressure build-up, gas leakage, gas pathways along drifts, etc); during this early phase, no radionuclide transport is expected.
Effects of engineering and stray materials	Effects of concrete bottom plate, stray materials, bentonite pellets.	Effects of steel rings, rock bolts, steel feet, water/gas evacuation pipes, grouting, spray and drip shields, cement.
Supercontainer and other structural components within the deposition drifts		
Materials, geometry, properties	N/A	
Steel corrosion and formation of corrosion products	N/A	For the expected steel corrosion rate, complete conversion to oxidised species occurs within a few thousand years.
Gas generation by anaerobic corrosion of steel	N/A	Gas generation rates are significant although the overall amount of gas produced is moderate; for the effects of gas, see buffer.

Effects of volume expansion (magnetite formation)	N/A	Volume expansion of corrosion products may increase buffer density and swelling pressure.
Ion release to bentonite porewater	N/A	Leads to iron/bentonite interaction.
Effect of supercontainer on water flow paths along the periphery of the drift	N/A	The physical properties of the corroded supercontainer have not been evaluated. Although the porosity and hydraulic conductivity of the corrosion products may be low, the possibility that fracturing could lead to the formation of pathways for water flow and advective transport cannot currently be excluded. Selected radionuclide transport calculation cases cover the case of a disturbed buffer/rock interface due to the presence of iron corrosion products in contact with bentonite.
Displacement of super-container/buffer by swelling of distance blocks	N/A	See buffer.
Breaching of supercontainer shells by bentonite swelling	N/A	The supercontainer shell may be breached by the different forces due to bentonite swelling acting inside and outside the supercontainer shell (secondary effect, because the supercontainer has no safety function).
Deposition drift, central tunnel, access tunnel, shafts, boreholes		
A major difference is in the geometry and backfilling of the KBS-3H deposition drifts compared with the KBS-3V deposition tunnels. In KBS-3H, supercontainers are emplaced along relatively narrow deposition drifts, separated by compacted bentonite distance blocks. In KBS-3V, deposition holes are bored from relatively large diameter deposition tunnels, backfilled with swelling clay or clay/crushed rock mixture.		
For other underground openings (access tunnel, shafts, boreholes) no major differences have been identified.		
Geosphere		
Gas transport, gas-induced porewater displacement	Relevant only for failed canisters.	Limited storage volume and transport capacity within deposition drift, combined with increased gas generation (rates and total amount). Gas dissolution/diffusion/advection in groundwater, gas pressure build-up, gas-induced porewa er displacement, capillary leakage. For tight canister sections: gas transport along drift (EDZ) to the next transmissive fracture, possibly involving reactivation of fractures in near-field rock, when minimal principal stress is exceeded.
Transmissive fractures and flow conditions	The selection of deposition hole locations is more flexible than in KBS-3H because rock sections with larger inflows can be rejected.	Local variations in groundwater flow conditions along the drift may lead to variable saturation time for the buffer along the drift (see Figure 6-3).
Mechanical stability of the drift/tunnel	High stresses at the mouth of deposition holes and at the top of backfill tunnel.	Lower rock stresses than in KBS-3V because the deposition drifts can be better adapted to the stress field.
Orientation of fractures	KBS-3V is more sensitive to sub-horizontal than to sub-horizontal fractures with respect to potential damage to the engineered barrier system by rock shear.	KBS-3H is more sensitive to sub-vertical fractures with respect to potential damage to the engineered barrier system by rock shear.
Biosphere, human activities		
No major differences identified		

3.6 Complementary evaluations of safety

It is required in Finnish regulations that "*The importance to safety of such scenarios that cannot reasonably be assessed by means of quantitative analyses shall be examined by means of complementary considerations. They may include e.g. bounding analyses by simplified methods, comparisons with natural analogues or observations of the geological history of the disposal site. The significance of such considerations grows as the assessment period of interest increases, and the judgement of safety beyond one million years can mainly be based on the complementary considerations*" /STUK 2001/.

The Complementary Evaluations of Safety Report supports the safety studies for the KBS-3H repository at the Olkiluoto site by bringing together the evidence and arguments which are complementary to the quantitative safety assessment. It brings together evidence on a range of scales such as:

- general supporting arguments for the concept of geological disposal,
- support for the robustness of the KBS-3H repository by comparison with similar concepts,
- support for the rigour and completeness of the quantitative assessment by comparison with those made elsewhere on similar concepts and/or similar host rocks,
- support from natural analogues for the choices of data used in the assessment,
- support from natural analogues that the repository system is founded on sound understanding of the behaviour the components over the long timescales required,
- support from complementary evaluations of the calculated releases which indicate the relative insignificance of the calculated doses and the low level of hazard implied.

Some of this evidence is described in Chapter 10 and in other sections of the present report.

Issues relating to quality assurance of both the system and the safety assessment studies are also discussed in the Complementary Evaluations Report (see below).

3.7 Uncertainty management and quality assurance of data, models and calculations

All computer codes used in Posiva's safety assessments are developed according to a quality assurance procedure and verified by comparison with analytical solutions, alternative codes and experimental data. Confidence in the modelling results is enhanced by means of the simulation of experiments and of natural analogue data. Posiva participates or has participated in international model validation studies such as the INTRAVAL (International Project to Study Validation of Geosphere Transport Models) project, which ended about 10 years ago, when the need for site-specific validation studies emerged. Currently, benchmarking studies are used to compare different codes to the same system. For instance, the VTT-developed code REPCOM has been verified against PORFLOW by /Nordman and Vieno 2003/. Posiva is also indirectly participating in international integration groups, such as DECOVALEX (DEvelopment of COupled models and their VALidation against EXperiments in nuclear waste isolation) and NF-PRO (Near field Processes) aimed at using different tools for modelling coupled thermo-mechanical-hydrological and chemical processes in deep geological disposal systems.

Input data used in the KBS-3H long-term safety studies has been collected in Appendix A of the KBS-3H Process Report /Gribi et al. 2007/ along with main sources of data and assumptions. The models, datasets and computer codes used in the radionuclide release and transport calculations, including the use of realistic and conservative model assumptions and data where there are uncertainties, are described in detail in the Radionuclide Transport Report and are also mentioned briefly in Section 9.4 of the present report. Data for analysing assessment cases are based on the preliminary design and data available at the time of writing the present report.

The motivation for and plausibility (or conservatism) of selected parameter values and model assumptions used have been reported as much as possible given the time constraints. The discussion is, however, often limited and largely qualitative. In future, Posiva plans to produce a data report common to KBS-3H and 3V, similar to the SR-Can data report /SKB 2006b/, that contains fuller discussion of all data used.

For modelling near-field release and transport, extensive use has been made of SR-Can parameter values and model assumptions, complemented where these are found to be inapplicable by "expert judgement". In the case of geosphere transport modelling, the modelling approach and parameter values used are based largely on TILA-99, although more recent developments in the understanding of the Olkiluoto site are used to provide additional support for the parameter values selected (e.g. in terms of their conservatism).

An important outcome of the safety assessment is an identification of remaining issues and uncertainties that need to be considered in future studies. These issues and uncertainties are described in Chapter 11.

4 KBS-3H design, safety concept and safety functions

4.1 The current reference design

The current reference design for the main engineered components of a KBS-3H repository is illustrated in Figure 4-1. The spent fuel is encapsulated in copper canisters with cast iron inserts. Each canister, with a surrounding partly saturated layer of bentonite clay, is placed in a perforated steel cylinder prior to emplacement and the entire assembly is called the supercontainer. The supercontainers are positioned along parallel, 100 to 300 m long deposition drifts. In order to provide drainage during the operational period, the drifts will have a shallow dip towards the tunnels from which they are bored. The supercontainers are supported by steel feet to leave an annular gap to the drift wall (about 4 cm in the current reference design). Adjacent supercontainers in the drift are separated from each other by bentonite distance blocks. Void spaces around the supercontainers and distance blocks will become filled with bentonite as the drift saturates and the bentonite swells, although the rate at which this occurs may vary considerably along the drift due the heterogeneity of the rock and the variability of water inflow, as discussed at length in Section 5.5 of the Evolution Report and summarised in Section 6.2.4 of the present report. The bentonite material emplaced as part of the supercontainers together with the bentonite distance blocks are jointly termed the buffer.

The main geometrical dimensions of these components are shown in Figure 4-2.

At the beginning of each deposition drift, adjacent to its intersection with the central tunnel (before the drift end plug), a 15-metre long, wider section of the tunnel (with a 50 m^2 cross section) hosts the deposition equipment for supercontainers and distance blocks during repository operations. This section is called "deposition niche".

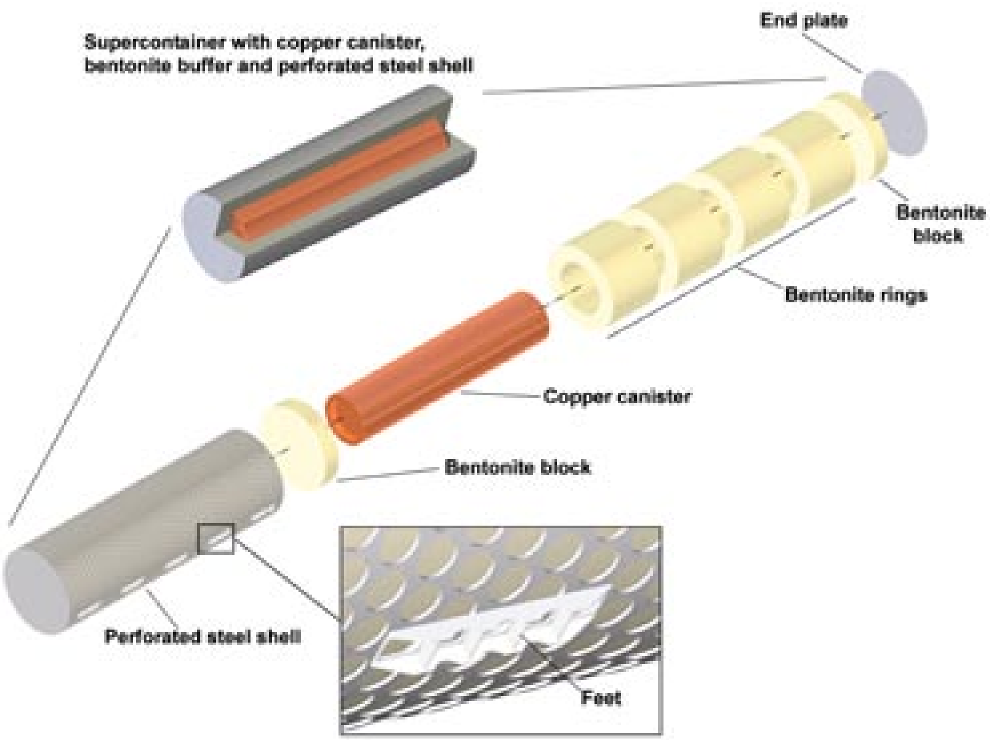


Figure 4-1. *The supercontainer with buffer and copper canister.*

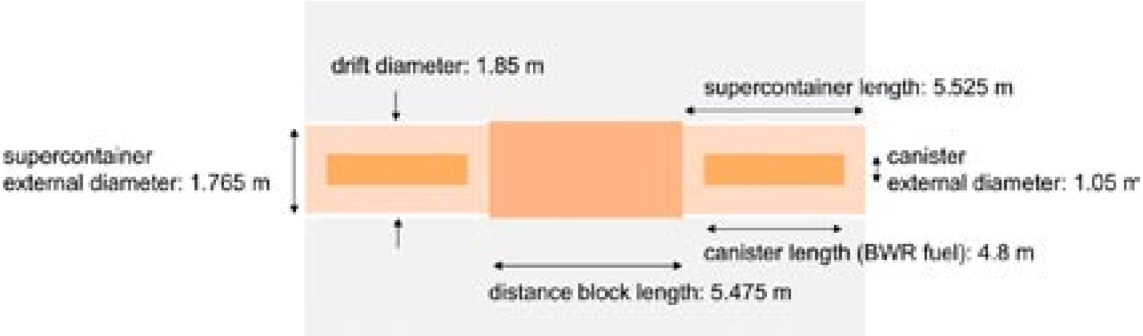


Figure 4-2. Illustration of asection of a KBS-3H deposition drift with two supercontainers separated by a distance block. The 5,475 m distance block length is for the reference fuel for the present safety studies (BWR spent fuel from the Olkiluoto 1 and 2 reactors).

Two broad realisations of KBS-3H design variants are currently being developed in parallel – the Basic Design, which is the current reference design option as outlined above, and the Drainage, Artificial Watering and air Evacuation (DAWE) design variant /Autio 2007, Autio et al. 2007/. The description of the system and its evolution in the following chapter is for the current reference design. Differences between the design options regarding the evolution of the repository principally affect the early evolution phase, prior to any possible release of radionuclides. The final saturated state of the repository is essentially the same whichever option is implemented.

4.2 Groundwater control and compartmentalisation of the drifts

During the saturation of the repository, high hydraulic pressure gradients and gradients in buffer swelling pressure may develop along the drifts, which could potentially lead to detrimental phenomena such as piping and erosion of the buffer and displacement of the distance blocks and supercontainers (see Section 7.1.2 for a discussion of these issues). Groundwater control measures in the form of pre- or post-grouting to reduce inflow to the drifts will be implemented for reasons of engineering practicality and operational safety, and to limit the possibility of detrimental phenomena such as those mentioned above. Currently, the possibility of using a large-scale post-grouting device (called Mega-Packer) to inject Silica Sol (colloidal silica), or other types of grout (low-pH cement) into transmissive fractures is being investigated /Autio et al. 2008/.

The performance of grout in reducing inflows is currently uncertain, and grout cannot be relied upon in the long term to reduce groundwater flow around the drifts. Thus, irrespective of any grouting that is undertaken, transmissive fractures intersecting the drift may render particular sections unsuitable for the emplacement of supercontainers and distance blocks. In the current reference design, additional “filling blocks” are emplaced in those ~ 10 m long drift sections where the total inflow to the section exceeds about 0.1 litres per minute, but excluding those higher inflow drift sections where inflow exceeds about 1 litre per minute even after grouting[9]. The maximum allowable inflow of 1 litre per minute is thus higher in the case of filling blocks compared with the 0.1 litres per minute allowed for distance blocks. This is because of the different functions of these two components. The distance blocks should prevent significant water flow by piping between adjacent supercontainer drift sections during saturation of the drift, which could otherwise lead to buffer erosion, as described in Section 7.1.2. The limit of

[9] Note that fractures giving inflows of less than 0.1 litres per minute may not be amenable to further flow reduction by grouting on account of their small apertures.

0.1 litres per minute is related to this requirement. The filling blocks, on the other hand, are not used to separate adjacent supercontainers and so the prevention of piping is not a primary consideration in deciding where they can be emplaced. There is, however, a requirement to avoid erosion of these blocks by water flowing around the drift through intersecting transmissive fractures and erosion. The relevant inflow criterion is expected to be higher, although the present choice of 1 litre per minute is ~~a~~ preliminary and somewhat arbitrary value that may be updated in view of future studies and possible design changes. Compartment plugs are used to seal off drift sections where inflows are higher than 1 litre per minute after grouting, thus dividing the drift into compartments. The design of the compartment plugs is still under development and demonstration tests are planned to be performed at the Äspö Hard Rock Laboratory. In the current provisional design, the compartment plugs are steel structures, as described in Section 4.3 of /Autio et al. 2007/. Each compartment plug is required to stay in place under the applied loads (no significant displacement allowed) until the next compartment is filled and a further compartment plug or drift end plug installed. Drift end plugs – low-pH concrete bulkheads in the current reference design – are placed at the ends of the drifts. The drift end plugs are designed to stay in place under the applied loads until the adjoining transport tunnels are backfilled. Steel fixing rings are installed, where necessary, along the drift to avoid displacement of the distance blocks and supercontainers prior to the installation of compartment and drift end plugs. These main features of a KBS-3H drift are shown in Figure 4-3.

According to current understanding and hydrogeological modelling of the site, a 300-metre long KBS-3H drift at Olkiluoto contains, on average, two compartments, 3–4 filling blocks, and 22 to 23 supercontainers /Lanyon and Marschall 2006/. On average, 17% of the drift is unusable due to high water inflow. A higher figure of 25% is tentatively and conservatively assumed in the layout adaptation (see Section 5.2). In this case, the drift contains, on average, 17–18 supercontainers. This takes into account the possibility that some relatively tight fractures that have the potential to undergo shear movements sufficiently large to damage the canisters in the event of a large earthquake are identified and avoided. The impact of earthquakes and rock shear on canister integrity is discussed in Section 7.2.5 in the context of future glaciation and post-glacial earthquakes.

4.3 Safety concept and safety functions

Many aspects of evolution of KBS-3V and KBS-3H repositories are expected to be the same or very similar. Furthermore, the essential elements of the KBS-3H safety concept – i.e. the conceptualisation of how the proposed system provides safety – are shared with KBS-3V. This common safety concept is illustrated in Figure 4-4, which shows the primary roles and relationships between the different technical components of the disposal systems.

Figure 4-3. *Illustration of a generic KBS-3H drift showing one canister in copper colour for better visualisation. At one end of the drift, a wider area (deposition niche) hosts the deposition equipment while the other end of the drift is closed off.*

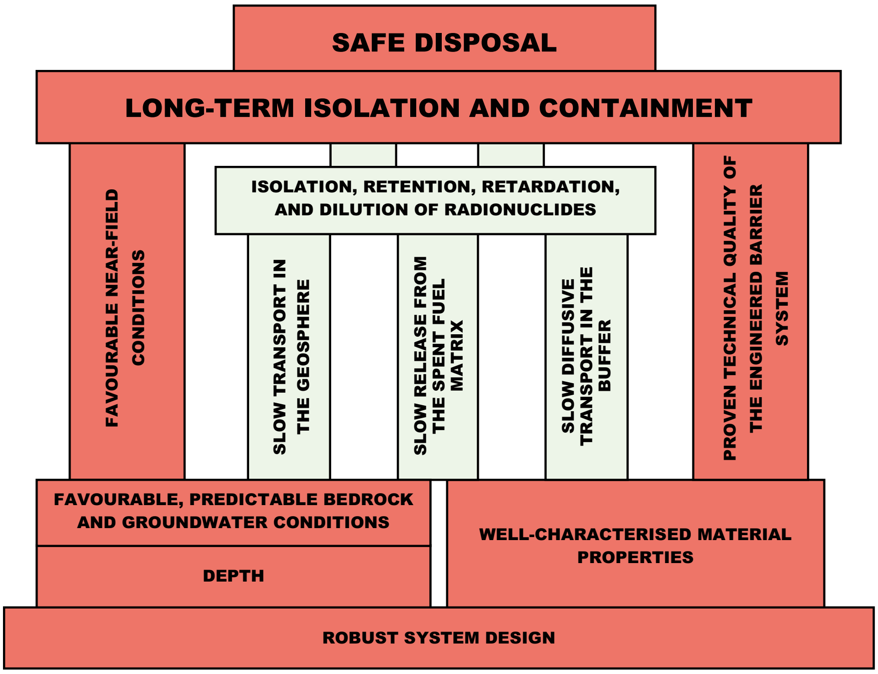


Figure 4-4. *Outline of the safety concept for a KBS-3 type repository for spent fuel in crystalline bedrock. Red pillars link characteristics of the disposal system to other characteristics on which they primarily depend. Green boxes and pillars indicate secondary characteristics and dependencies /Posiva 2006/.*

The canister, the buffer (i.e. the bentonite material originally inside the supercontainers, together with the distance blocks) and the host rock are the main KBS-3H system components that together ensure isolation of the spent fuel and containment of radionuclides. Each of these components performs a number of safety functions. Following SR-Can /SKB 2006a/, a safety function is defined as a qualitative role through which a repository component contributes to safety. Other KBS-3H system components, including the steel supercontainers, fixing rings, compartment and drift end plugs and other structural materials, are not assigned safety functions. They are, however, designed to be compatible with, and support the safety functions of, the canister, the buffer and the host rock.

The main safety function of the canister, which is common to KBS-3H and KBS-3V, is to ensure a prolonged period of complete containment of radionuclides. As long as its copper shell is not breached, a canister will provide complete containment of radionuclides, and the spent fuel will interact with the environment only by means of heat generation and low level gamma and neutron radiation penetrating through the canister walls. This safety function rests first and foremost on the mechanical strength of the canister insert and the corrosion resistance of the copper surrounding it. In the current reference design, the canisters have a design lifetime of at least 100,000 years.

This means that the canisters are designed to maintain their integrity taking into account the processes and events that are considered likely to take place in the repository over a design basis period of 100,000 years. It does not exclude the possibility that canister integrity will be retained significantly beyond the design basis period, nor that less likely, extreme conditions will give rise to earlier canister failures, and these possibilities must be considered in safety assessment. The terminology is similar to that used in the reactor safety area: a design basis is defined to reflect the most likely conditions for the reactor but safety assessment addresses less likely situations as well. If the copper shell is breached, then a canister is considered to have failed, even though it may continue to offer some resistance to the ingress of water and the release of radionuclides for a significant period thereafter.

Safety functions of the buffer include (a), protection of the canisters, and (b), limitation and retardation of radionuclide releases in the event of canister failure. These safety functions are also common to KBS-3V and KBS-3H. A final safety function of the KBS-3H buffer (or, more specifically, the distance blocks) is, (c), in addition, to separate the supercontainers hydraulically one from another, thus preventing the possibility of preferential pathways for flow and advective transport along the drift. The buffer is designed (i), to be sufficiently impermeable, once saturated, that the movement of water is insignificant and diffusion is the dominant radionuclide transport mechanism, (ii), to have a sufficiently fine pore structure such that microbes and colloids are immobile (filtered) and microbe- or colloid-facilitated radionuclide transport will not occur in the event of canister failure and (iii), to provide tight interfaces with the host rock within a reasonable time. Bentonite clay also has a self-healing capability, which means that any potential advective pathways for flow and transport that may arise, for example, as a result of piping and erosion, sudden rock movements or the release of gas formed in a damaged canister should be rapidly closed.

The safety functions of the host rock are also common to KBS-3H and KBS-3V. They are (a), to isolate the spent fuel from the biosphere (b), to provide favourable and predictable mechanical, geochemical and hydrogeological conditions for the engineered barriers, protecting them from potentially detrimental processes taking place above and near the ground surface, and (c), to limit and retard inflow to and release of harmful substances[10] from the repository. In order to ensure the isolation of the repository and to provide the required favourable host rock properties, it is evident that geological stability, with low rates of tectonic uplift[11] and tectonic activity, of the site is a pre-requisite. The location of a site in the Fennoscandian shield and especially in Finland is advantageous with respect to the stability of the geosphere /Marcos et al. 2007/.

4.4 Safety function indicators and criteria

To assess the performance and safety of a KBS-3H or KBS-3V repository, it is necessary to assess the conditions under which the identified safety functions will operate as intended, and the conditions under which they will fail, or operate with reduced effectiveness. Following the methodology adopted in the Swedish SR-Can safety assessment /SKB 2006a/, the present safety assessment makes use of the concept of safety function indicators and associated criteria. One or more safety function indicators are assigned to each safety function. A safety function indicator is a measurable or calculable property of the system that is critical to a safety function being fulfilled. If the safety function indicators meet certain criteria, then the safety functions can be assumed to be provided. If, however, plausible situations can be identified where the criteria for one or more safety function indicators are not fulfilled, then the consequences of loss or degraded performance of the corresponding safety function must be evaluated in the safety assessment.

Safety function indicators and criteria for the canister, the buffer and the host rock are given in Tables 4-1 to 4-3. The rationale behind individual criteria is discussed in Section 2.5 of the Evolution Report /Smith et al. 2007b/ and, at greater length, in SR-Can /SKB 2006a/, from where the majority of criteria are taken. It should be noted that the criterion given in Table 4-2 that there is a negligible impact on the rheological and hydraulic properties of the buffer due to mineral alteration subsumes the SR-Can criterion for a Swedish KBS-3V repository that buffer temperature remains below 100°C. The present criterion takes account of the concern that the

[10] Including the chemically toxic components of spent fuel, as discussed in the Complementary Evaluations Report /Neall et al. 2007/.

[11] It should be noted that tectonic uplift is not the same as post-glacial (isostatic) uplift. The apparent uplift rate, which is the rate of isostatic uplift minus the eustatic component due to sea-level change associated with the changing shapes of the sea basins, is 6 mm per year (this does not include the impact of global sea level change). The land uplift rate is expected to vary little over the next few centuries, but will decrease significantly within the next few thousand years /Ruosteenoja 2003/.

buffer of a KBS-3H repository may be more affected by certain chemical interactions, and particularly those between the corrosion products of steel components external to the canisters and bentonite and those between cementitious materials and bentonite, than is the case for a KBS-3V repository (Chapter 7).

It is emphasised that, if there are plausible situations where one or more of the criteria for safety function indicators are not satisfied, this does not imply that the system as a whole is unsafe. Such situations must, however, be carefully analysed, for example by means of radionuclide release and transport calculations (Chapter 9).

Table 4-1. Safety function indicators and criteria for the canister (after Figure 7-2 of /SKB 2006a/).

Safety function indicator	Criterion	Rationale
Minimum copper thickness	> 0 mm	Zero copper thickness anywhere on the copper surface would allow relatively rapid water ingress to the canister interior and radionuclide release
Isostatic pressure on canister	< pressure for isostatic collapse (varies between canisters, but probability of collapse at 44 MPa is vanishingly small)	An isostatic pressure on the canister greater than 44 MPa would imply a more significant possibility of failure due to isostatic collapse
Shear stress on canister	< rupture limit	A shear stress on the canister greater than the rupture limit would imply failure due to rupture

Table 4-2. Safety function indicators and criteria for the buffer (adapted for KBS-3H from Figure 7-2 of /SKB 2006a/).

Safety function indicator	Criterion	Rationale
Bulk hydraulic conductivity	**$< 10^{-12}$ m s^{-1}**	**Avoid advective transport in buffer**
Swelling pressure at drift wall	> 1 MPa	Ensure tightness, self sealing
Swelling pressure in bulk of buffer	> 2 MPa	Prevent significant microbial activity
	> 0.2 MPa[1]	Prevent canister sinking
Saturated density	> 1,650 kg m^{-3}	Prevent colloid-facilitated radionuclide transport
	< 2,050 kg m^{-3}	Ensure protection of canister against rock shear
Mineralogical composition	No changes resulting in significant perturbations to the rheological and hydraulic properties of the buffer (e.g. from iron or cement interactions or related to temperature)	See main text
Minimum buffer temperature	> –5°C	Avoid freezing

[1] Although developed for KBS-3V, this criterion is also expected to be applicable to KBS-3H, and is likely to be more conservative for this concept since, in KBS-3H, the weight of the canister is distributed over a larger horizontal area compared with KBS-3V

Table 4-3. Safety function indicators and criteria for the host rock (adapted for KBS-3H from Figure 7-2 of /SKB 2006a/).

Safety function indicator	Criterion	Rationale
Redox conditions	No dissolved oxygen	The presence of measurable O_2 would imply oxidising conditions
Minimum ionic strength	Total divalent cation concentration > 10^{-3} M	Avoid buffer erosion
Minimum pH or maximum chloride concentration	$pH^{GW} > 4$ or $[Cl^-]^{GW} < 3$ M	Avoid chloride corrosion of canister
Limited alkalinity	$pH^{GW} < 11$	Avoid dissolution of buffer smectite
Limited salinity (expressed in terms of total dissolved solids, TDS)	[NaCl] < 100 g/l (or other compositions of equivalent ionic strength)	Avoid detrimental effects, in particular on swelling pressure of buffer and distance block
Limited concentration of detrimental agents for buffer, distance block and canister	Applies to HS^-, K^+ and Fe(II)/Fe(III). The lower the better (no quantitative criterion)	Avoid canister corrosion by sulphide, avoid illitisation (K^+) and chloritisation (Fe) of buffer and distance block
Limited rock shear at canister/distance block locations in deposition drift	< 10 cm	Avoid canister failure due to rock shear in deposition drift

5 Initial conditions

5.1 The Olkiluoto site and host rock

5.1.1 Reporting of site investigations

The following sections give a brief description of the Olkiluoto site, the location of which is shown in Figure 5-1, which also indicates the Swedish sites at Forsmark and Laxemar that were the subject of SR-Can. The description, which focuses largely on those aspects most relevant to the present safety assessment, is based on the comprehensive overview of investigations carried out at Olkiluoto up to the year 2005, most recently compiled into Site Report 2006 /Andersson et al. 2007/. This description is, in turn, an update of the descriptions provided in the Baseline Report /Posiva 2003 and 2004/ version of the site description /Posiva 2005/. An extensive summary of the site description is provided in Chapter 11 of /Andersson et al. 2007/. The reader is referred to /Andersson et al. 2007/ and to the description of the site given in Section 2.2 of the Evolution Report for references supporting the statements given in the following site description.

Figure 5-1. *The location of the Olkiluoto site, which is the subject of the present study, and the Forsmark and Laxemar sites, which were the subject of SR-Can.*

5.1.2 Surface conditions and bedrock geology

Olkiluoto is a relatively flat island with an average height of 5 m above sea level, covered by forest and shoreline vegetation and surrounded by shallow sea, with a depth mainly less than 12 m within 2 km of the current shoreline. The overburden, both onshore and offshore, is mostly till with a thickness generally between 2 and 4 m. The apparent rate of land uplift is significant at 6 mm per year, mainly due to isostatic adjustment of the bedrock /Haapanen et al. 2007/. Thus, the elevation of the island relative to sea level is continuously changing.

The groundwater table follows the form of the surface topography and is mainly 0 to 2 m below surface, with some exceptions. Olkiluoto Island forms its own hydrological unit; the surface waters flow directly into the sea /Lahdenperä et al. 2005, Posiva 2005/. Infiltration of surface water is currently being investigated. Current (and provisional) estimates are approximately 1–2% of the annual precipitation. The evolution of surface conditions and the local ecosystem are described in Section 6.2.1 and, in more detail, in the Terrain and Ecosystems Development Model report /Ikonen et al. 2007/.

The bedrock at Olkiluoto belongs to the Svecofennian domain of Southern Finland and comprises a range of high-grade metamorphic and igneous rocks. The metamorphic rocks include various migmatitic gneisses and homogeneous, banded or only weakly migmatised gneisses, such as mica gneisses, quartz gneisses, mafic gneisses and tonalitic-granodioritic-granitic gneisses. The igneous rocks comprise abundant pegmatitic gneisses and sporadic narrow diabase dykes.

According to Finnish regulations, the importance to safety of a substantial rock movement occurring in the environs of the repository should be considered Seismic activity in the Olkiluoto region is currently low (see, e.g. /La Pointe and Hermanson 2002, Enescu et al. 2003, Saari 2006/). GPS and seismic measurements at Olkiluoto show negligible rock movements /Andersson et al. 2007/. As discussed, for example, in Section 2.2.2 of the Evolution Report, the lack of indications of recent significant seismic activity at Olkiluoto, together with indications of post-glacial reactivation in the past, suggest that any major seismic activity in the future is also likely to occur with the greatest frequency following glaciations /Andersson et al. 2007/, although infrequent but significant seismic events during inter-glacial periods cannot be excluded. Ongoing geological characterisation and modelling will give more insight into the characteristics of deformation zones at the site and their potential to host future large post-glacial earthquakes.

5.1.3 Rock fracturing and groundwater flow

The Olkiluoto bedrock contains a network of fractures and fracture zones. The frequency, spatial distribution, size distribution, shape and orientation of these structures affect both the hydraulic and mechanical properties of the rock. The fracture zones often constitute dominant paths for groundwater flow and their size also affects the scale of any rock shear movements that take place in the event of earthquakes. Hydrothermal alteration related to the late stages of metamorphism has occurred in certain domains in the rock mass, and this affects its strength and transport properties.

Up till now, the focus of the hydrogeological modelling of the Olkiluoto site has been on identifying and characterising the major hydraulically active deformation zones, whereas the rock masses between these zones have been given average hydraulic properties. Detailed data on the scale of individual fractures also exist, although so far the analyses of these data have focused on those fractures reflecting likely conditions in the deposition drifts /Hellä et al. 2006/. According to these analyses, transmissive fractures at relevant depths, especially those with transmissivities higher than 10^{-8} m^2 s^{-1}, are concentrated mainly in local zones of abundant fracturing. Fractures with lower transmissivities occur outside these zones, but also tend to form clusters /Hellä et al. 2006/. The rock matrix between fractures has an average porosity of 0.14% /Autio et al. 2003/ and a low hydraulic conductivity so that water fluxes through it are negligible compared to that through fractures.

Calcite and a range of clay minerals (illite, smectite, kaolinite, vermiculite and chlorite) make up most of the fracture filling. Pyrite is also abundant, mainly as coatings on calcite grains. Pyrite has been observed in all boreholes studied so far at the site. These fracture fillings play an important role in the hydrogeochemical conditions at Olkiluoto and their evolution, as discussed in the Process Report /Gribi et al. 2007/.

5.1.4 Groundwater composition

The groundwater composition over the depth range 0 to 1,000 m at Olkiluoto is characterised by a significant range in salinity (see, e.g. Figure 11-8 in /Andersson et al. 2007/). Fresh groundwater with low total dissolved solids (TDS less than about 1 g/l) is found only at shallow depths, in the uppermost tens of metres. Brackish groundwater, with TDS up to 10 g/l dominates at depths between 30 m and about 400 m. Saline groundwaters (TDS > 10 g/l) dominate at still greater depths. The current salinity of groundwater at repository depth (400 to 500 m below ground) ranges from 10 to 20 g per litre TDS /Andersson et al. 2007/. Chloride is normally the dominant anion in all bedrock groundwaters. Near-surface groundwater is also rich in dissolved carbonate and groundwater at depths between about 100 and 300 m is characterised by high sulphate concentrations. Both carbonate and sulphate concentrations decrease significantly at greater depths. Sodium and calcium dominate as main cations in all groundwaters, and magnesium is also notably enriched in sulphate-rich waters.

Dissolved nitrogen and methane content is high in Olkiluoto groundwater, nitrogen being the dominant dissolved gas in the upper 300 m with methane being dominant at greater depths. Redox conditions deep underground are reducing, and are buffered by a range of redox processes, with microbes playing an important role. The anaerobic reduction of sulphate to sulphide at the interface where sulphate-rich marine derived brackish groundwater is comes into contact with deeper, methane-rich saline groundwater results in an abundance of sulphide in the groundwater at repository depth, the highest measured sulphide concentration being 12 mg per litre. The sulphate content in the sulphate-rich brackish groundwater is at maximum about 500 mg/l, decreasing to a negligible concentration in the methane-rich saline groundwater /Pitkänen et al. 2004/. In a repository, sulphide may diffuse through the buffer and react with and corrode the copper canister. Sulphide ions thus play an important role in considerations of canister lifetime. As noted in Chapter 11, the role of methane in the reduction of sulphate to sulphide, its evolution over time and its impact on canister lifetime are issues requiring further investigation.

The pH conditions in the deep aquifer system at Olkiluoto are well buffered by the presence of abundant carbonate and clay minerals found in fracture fillings. The pH values at relevant depths are generally in the range 7.5–8.2 /Pitkänen et al. 2004/.

Below about 300 m depth, studies of methane and of isotopic composition indicate that the deep stable groundwater system has not been disturbed by glacial and post-glacial transients and that neither oxidising glacial meltwater nor marine water have affected the composition of this deeper system.

5.1.5 Rock stress

The repository drifts will be aligned as much as possible with the direction of the maximal horizontal stress for reasons of mechanical stability (Section 5.2). Regional data indicate that the mean orientation of the maximum horizontal stress is roughly E-W, but the data display a large scatter, so that it is currently uncertain whether the stress orientation at the site differs from the mean regional orientation. At 500 m, the maximal horizontal stress is estimated to be between 15 and 31 MPa and the minimum horizontal stress is estimated to be in the range 10 to 18 MPa. The vertical stress is estimated to be between 7 and 15 MPa at 500 m. The major principal stress is subhorizontally orientated, and is thus slightly larger in magnitude than the maximum horizontal stress. The other two principal stress components vary significantly in magnitude and orientation between the different measurement locations, indicating the need to relate the stress field to geological structure and to conduct associated numerical analyses.

5.1.6 Post-glacial adjustment

During the last glaciation, weight of the ice mass against the viscous mantle caused the Earth's crust to sink some hundreds of metres. As the ice sheet started to melt about 13,500–10,300 years ago, the crust started to rise. The Earth's crust at Olkiluoto is currently still in the process of returning to its position of isostatic equilibrium. The most obvious consequences of postglacial adjustment in Fennoscandia are the land uplift along both sides of the northern part of the Baltic Sea and the concomitant retreat of the shoreline. Currently, the rate of isostatic post-glacial uplift at the site is estimated to be 6.8 mm per year /Johansson et al. 2002, Kahma et al. 2001, Eronen et al. 1995/. The apparent uplift rate, which is the rate of isostatic uplift minus the eustatic component due to sea-level change associated with the changing shapes of the sea basins, is 6 mm per year (this does not include the impact of global sea level change). The land uplift rate is expected to vary little over the next few centuries, but will decrease significantly within the next few thousand years /Mäkiaho 2005, Påsse 1996, 1997, 1998/.

5.1.7 Impact of repository excavation

The preceding sections describe the undisturbed hydrogeological and geochemical conditions at Olkiluoto representing the baseline conditions prior to starting of excavation of ONKALO. Excavation of ONKALO and of the repository tunnel and drift system will, however, significantly perturb these conditions, causing a transient drawdown of the water table and an associated reduction in the hydrostatic pressure at repository depth /Pastina and Hellä 2006/. Grouting of fractures in the rock is, however, likely to be used (Section 4.2), and this will significantly reduce hydraulic disturbance caused by excavation.

Figure 5-2 shows the modelled evolution of the maximum salinity at repository depth (about 400 m below ground) and at 550 m below ground[12] from the beginning of repository operations (2020) until 100,000 years later /Pastina and Hellä 2006/.

Excavation is likely to cause a transient increase in the mixing of water types. In particular, there will be mixing of fresh water and brackish, sulphate-rich marine-derived waters from closer to the surface as well as of saline water from greater depths. The mixing of the waters from closer to the surface may induce some changes in pH and Eh of the groundwaters, although buffering by the host rock should prevent significant Eh and pH changes. Preliminary calculations have been carried out based on the estimation of the lifetime of fracture mineral buffers, such as calcite and pyrite in the fractures, against acid pH and oxygen containing infiltrating waters from the surface /Andersson et al. 2007/. Further field studies and more advanced modelling are planned. At the time of emplacement of the first canister, the water is expected to be diluted as compared with the undisturbed conditions prior to excavations. During the operational phase (100 years) the salinity (TDS, total dissolved solids) may rise from a maximum of 12 g per litre at the time of emplacement of the first canister to a maximum of 25 g per litre in the vicinity of the excavations at repository level, about 400 m below ground (and from a maximum of 30 g per litre to a maximum of 47 g per litre at 550 m depth) according to groundwater flow simulations, depending on the extent of fracture grouting used /Vieno 2000, Vieno et al. 2003, Löfman 2005, Pastina and Hellä 2006/. This is due to upconing of deeper, saline waters while the repository is open.

In the post-closure period, groundwater salinity will again decrease towards values characteristic of undisturbed conditions, although calculations performed for a KBS-3V repository at Olkiluoto indicate that the thermal output from the spent fuel may cause an increased upward groundwater flow and increased salinity compared with undisturbed conditions for a period of up to several hundreds of years /Löfman 2005/.

[12] 550 m below ground is the lowest point in the ONKALO tunnel system, ONKALO being an underground characterisation facility that will also serve as an access route to the repository.

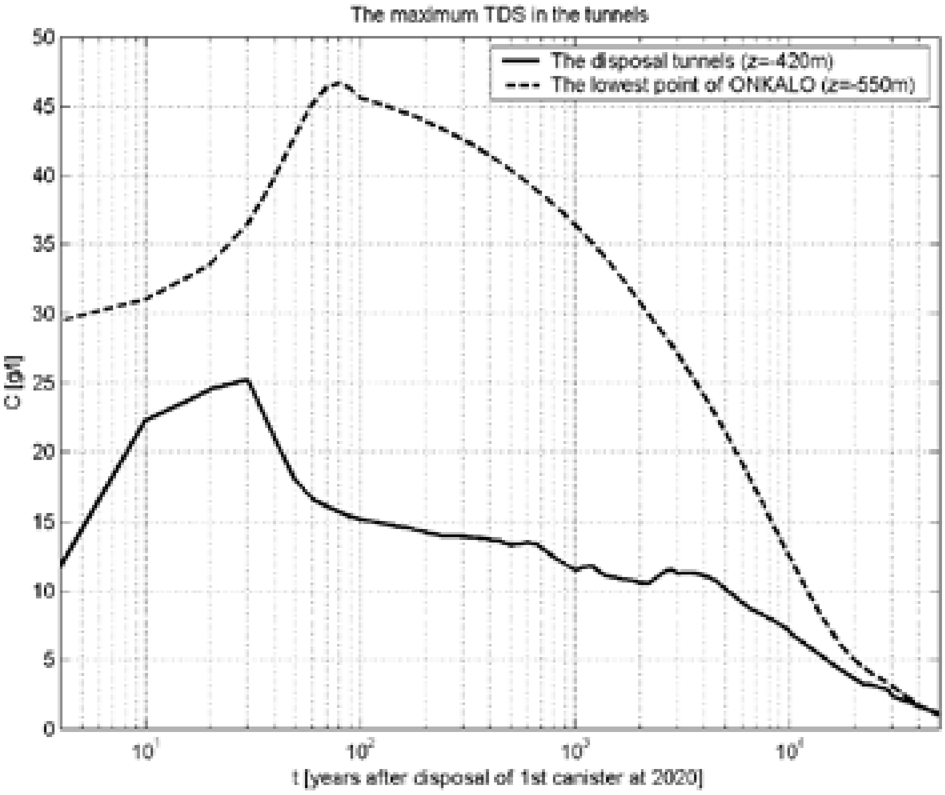


Figure 5-2. *Evolution of maximum salinity at the repository level (about 400 m below ground, solid line) expected from the beginning of repository operations (2020) onward. The starting salinity takes into account the effect of construction on salinity. The maximum salinity at the lowest point in the ONKALO tunnel system (550 m below ground, dotted line) is also shown for comparison. Salinity is expressed in g per litre of Total Dissolved Solids (TDS). The figure (after Figure 6-19 of /Pastina and Hellä 2006/) was developed for KBS-3V disposal tunnels, but is also expected to be applicable to KBS-3H deposition drifts.*

In addition to these transient hydrogeological and geochemical perturbations, excavation of underground openings will also give rise to irreversible mechanical disturbances[13] and, in particular, to the formation of excavation damaged zones (EDZs) around the openings. Such zones are potentially of significance to long-term safety because, if sufficiently conductive, they could create additional pathways for groundwater flow along the drifts. A drift excavation technique is being designed that should ensure that the degree of excavation disturbance is small. A description of the EDZ of a KBS-3H drift is given in Appendix D of /Johnson et al. 2005/. Current understanding is that the hydraulic significance of the EDZ is low, as assumed in the radionuclide release calculations described in the Chapter 9. EDZ properties are, however, subject to considerable uncertainties, being dependent on a variety of rock-specific and site-specific factors as well as on the repository layout and engineering techniques applied (see the discussion in Section 2.4 of /Johnson et al. 2005/), and will require further investigation in future project stages.

[13] An initial disturbance in terms of rock stress in the excavation disturbed zone (EdZ) could potentially lead to damage (fracturing) at later times.

5.2 Repository layout and depth

The layout of a repository at Olkiluoto will be adapted according to the locations of major deformation zones, which will be avoided when constructing deposition drifts. Two broad arrangements of deposition drifts are currently under consideration: 1) a one-storey facility with deposition drifts at a depth of approximately 400 metres (reference design) or 2) a two-storey facility with deposition drifts at depths of approximately 400 and 500 metres (alternative design). This present safety assessment addresses the evolution of a one-storey repository only, although any differences compared with the alternative design are expected to be minor.

The number, length and orientations of the drifts themselves will be constrained by a range of factors, including safety-related requirements regarding:

- the hydraulic properties of fractures intersecting the deposition drifts at canister and buffer emplacement locations,
- the maximum allowable temperature within the deposition drifts,
- the extent and hydraulic conductivities of the EDZs around the drifts, including the risk of rock spalling.

Efforts will be made to avoid fractures at canister emplacement locations that could undergo shear movements that could damage the canisters in the event of a large post-glacial earthquake, although the extent to which it will be possible to identify and avoid fractures capable of undergoing potentially damaging displacements is currently under investigation for both KBS-3V and KBS-3H (Chapter 11).

A preliminary layout for a one-storey facility based on the current bedrock model and the hydraulic, thermal and mechanical considerations described above is shown in Figure 5-3. In this layout, the repository is constructed in a single layer at a depth in the range 400 to 420 m below ground (note that the drifts are slightly inclined to the horizontal to facilitate drainage during construction and operation). The layout features 170 deposition drifts with an average length of 272 m. It is estimated by /Johansson et al. 2007/ that about 25% of the total drift length will be unusable for canister and buffer emplacement (see the discussion of groundwater control and compartmentalisation of the drifts in Section 4.2). Thus, the total "usable" drift length is about 34,700 m, which is sufficient for the approximately 3,000 canisters to be emplaced in the current KBS-3H reference design and canister pitch (Figure 4-2). The drifts are orientated at 120°, with their axes close to the direction of maximum horizontal stress.

This preliminary layout uses about 95% of the currently well-characterised and available bedrock resource[14] at Olkiluoto, which takes into account the major fracture zones shown in Figure 5-3, respect distances to investigation boreholes, the access tunnel to the ONKALO, the shoreline of Olkiluoto Island, and the boundaries of the current investigation area. Although the margin for uncertainties is small in this layout, it would, if necessary, be possible to increase this margin by extending the current investigation area (there is, for example, land to the east that is potentially available) or constructing the repository in multiple layers. It should be noted that the radionuclide release and transport calculations summarised in Chapter 9 and described in detail in the Radionuclide Transport Report consider releases from a single failed canister and transport along a single radionuclide transport path characterised by a given transport resistance. This simple modelling approach means that results are unaffected by whether a single or multiple-layer repository is assumed.

[14] For comparison, the current reference KBS-3V layout at Olkiluoto uses about 80% of the available bedrock resource /Johansson et al. 2007/.

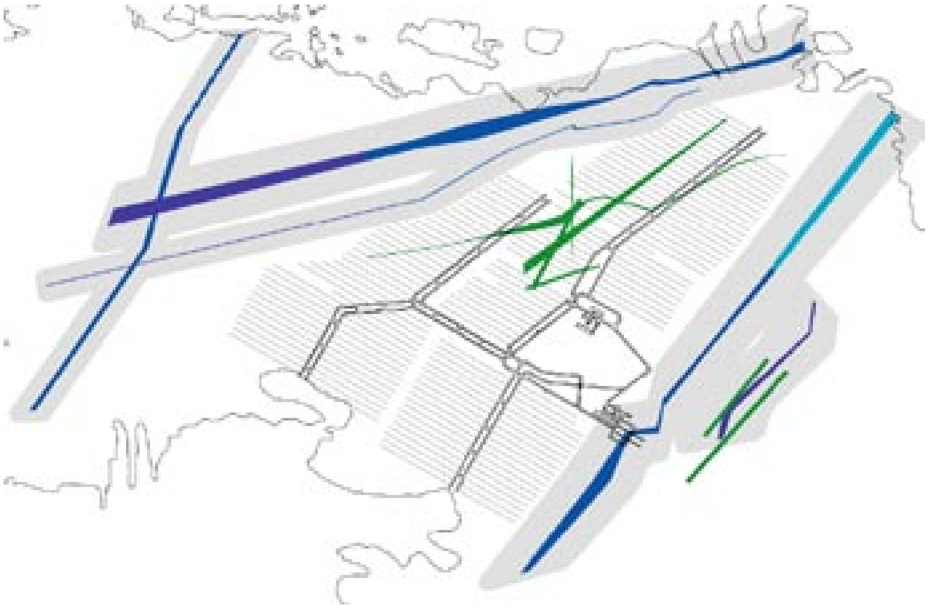


Figure 5-3. *Example layout adaptation of a KBS-3H repository at Olkiluoto at a depth of about 400 m below ground /after Johansson et al. 2007/. The layout is based on the Site Description 2006 /Andersson et al. 2007/ in which the hydraulic zones and their extension are taken into account. Grey areas indicate the respect distances to layout-determining fracture zones (shown in blue). Features shown in green are not considered to be layout determining, and may thus intersect the deposition drifts.*

5.3 The spent fuel

Spent fuel consists of cylindrical pellets of uranium dioxide, stacked in closed tubes (fuel rods) of zirconium alloy cladding (Zircaloy). Collections of fuel rods are integrated to form fuel assemblies, held together with spacers and plates. Activity inventories, heat production and other radioactive properties of Finnish spent fuel have been evaluated as functions of cooling time for different burnups, void history, and enrichment /Anttila 2005a/.

Irradiation of the fuel assemblies produces a large number of radionuclides. These radionuclides include those produced by the fission of uranium and plutonium in the fuel pellets (fission products), as well as activation products arising from neutron absorption. The majority of fission products and higher actinides in the fuel exist as a solid solution in the uranium dioxide matrix. However, some of the activation products, such as ^{14}C and ^{36}Cl, are present in both the fuel pellets and structural materials. Certain radionuclides are also enriched at grain boundaries in the fuel, at pellet cracks and in the fuel/sheath gap as a result of thermally driven segregation during irradiation of the fuel in the reactor, as illustrated in Figure 5-4.

For the purpose of safety assessment, the radionuclide inventory is assigned to three characteristic locations in a fuel rod, namely (i), the fuel matrix, (ii), the grain boundaries and gaps (instant release fraction), and (iii), structural materials (cladding).

The Finnish fuel considered in the present safety assessment includes:

- Loviisa 1-2: 700 canisters containing 1,020 tU of spent fuel
- Olkiluoto 1-2: 1,210 canisters containing 2,530 tU of spent fuel
- Olkiluoto 3: 930 canisters containing 1,980 tU of spent fuel
- Total: 2,840 canisters and 5,530 tU from the spent fuel

The fuel has an average burn-up of 37–39 MWd/kgU and a maximum burnup of 45 MWd/kgU for Loviisa and 50 MWd/kgU for Olkiluoto. Spent fuel inventory estimates have been based on maximum discharge burnup of 45 and 50 MWd/kgU. These estimates may change in the future along with the development of reactor load factors, fuel designs and burnup /Pastina and Hellä, 2006/.

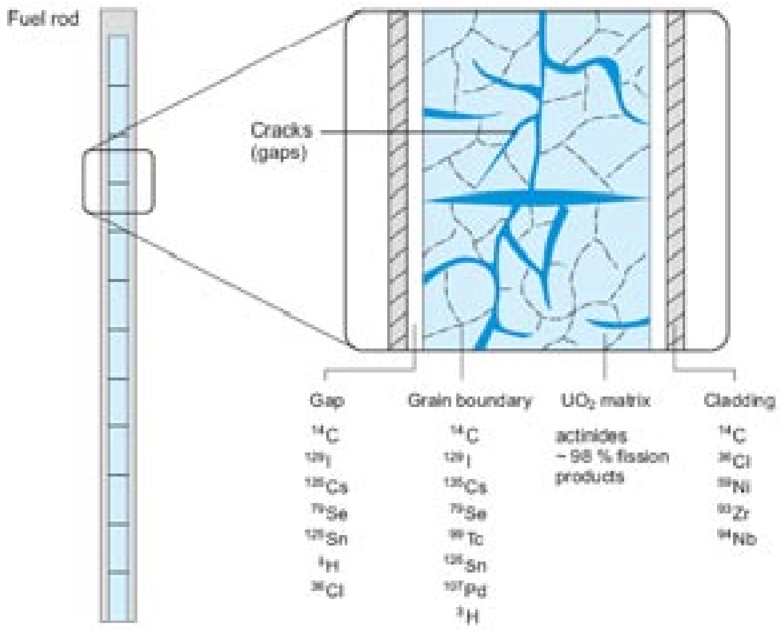


Figure 5-4. *Schematic illustration of the distribution of radionuclides within a fuel rod (from /Nagra 2002/, based on /Johnson and Tait 1997/).*

5.4 The canister

The spent fuel assemblies are sealed in copper canisters of which there are three versions, one for each reactor type in Finland. The main dimensions and masses of the three canister types are given in Table 5-1. The design and dimensioning of the canisters, as well as the manufacturing and encapsulation procedures and quality control, are described in detail in /Raiko 2005/.

Posiva's current reference canister sealing method is electron beam welding, with friction stir welding as an alternative option. SKB's reference sealing method is friction stir welding. Posiva is currently evaluating the advantages and disadvantages of the welding methods and has not yet made the final selection. In both methods, the seal location is on or within a few cm from the end-face of the canister so that no differences can be identified from the long-term safety point of view. These seals are the most likely locations for initial penetrating defects in the copper canisters. The likelihood of occurrence of initial penetrating defects is discussed further in Section 8.4

Table 5-1. Main dimensions and masses of canisters for different types of spent fuel /after Raiko 2005/.

	Loviisa 1-2 (VVER-440)	Olkiluoto 1-2 (BWR)	Olkiluoto 3 (EPR)
Outer diameter (m)	1.05	1.05	1.05
Height (m)	3.60	4.80	5.25
Thickness of copper cylinder (mm)	48	48	48
Thickness of copper lid and bottom (mm)	50	50	50
Total volume (m^3)	3.0	4.1	4.5
Fuel assemblies	12	12	4
Amount of spent fuel (tU)	1.4	2.2	2.1
Void space (m^3)	0.61	0.95	0.67
Mass of fuel assemblies (tonne)	2.6	3.6	3.1
Mass of iron and steel (tonne)	10.4	13.4	18.0
Mass of copper (tonne)	5.7	7.4	8.0
Total mass (tonne)	18.6	24.3	29.1

5.5 Drift temperature, air humidity and initial water inflow

The air temperature in open deposition drift sections will be controlled during operations. Following the sealing of a drift compartment, however, the heat generated by the fuel, the heat-transfer properties of the system components and the ambient rock temperature will determine subsequent temperature evolution. The ambient rock temperature at depths of 400 m and 500 m are 10.5°C and 12°C, respectively /Anttila et al. 1999/. The evolution of temperature in the drift during the operational period and at later times is described in Section 5.2 of the Evolution Report and summarised in Chapter 6 of the present report.

Water vapour in the air in the deposition drifts will originate from the rock and, during the operational period, from the water cushion system that is proposed for the deposition vehicle /Autio et al. 2007/. The humidity in the deposition drifts is expected to be close to 100% at the start of operations. Although evaporation from wet surfaces within the drift will tend to maintain high levels of humidity, this could be affected during the operational period by design features, such as the use of spray and drip shields, which would cover such surfaces, and will remain in place following sealing of a drift compartment. Humidity could also be reduced as bentonite is emplaced in the drifts and starts to take up moisture from the air.

Groundwater flow at Olkiluoto is concentrated in transmissive fractures. These are likely to intersect the deposition drifts at various locations and will lead to water inflow and saturation of gas-filled voids during the early, transient phase. Some water inflow to the drifts will occur at discrete locations due to intersections with transmissive fractures (or channels within those fractures), while some may be more dispersed by fractures in the EDZs surrounding the drifts. /Hellä et al. 2006/ have generated different realisations of the flow conditions in a deposition drift, based on the available field data, according to which the flow into 10 m drift intervals (corresponding to the length of a KBS-3H ”supercontainer unit” comprising one supercontainer and one distance block) is less than 0.1 litres per minute over about 85% percent of the drift length, 0.1 litres per minute inflow being the currently assumed maximum inflow criterion for a drift section to be suitable for canister and buffer emplacement (Section 4.2). There will, however, be considerable spatial variability in water inflow between different locations along the drift, with the tightest drift sections giving essentially no inflow of liquid water at all. The estimates of /Hellä et al. 2006/ may be revised as a result of the ongoing detailed site characterisation work at ONKALO and associated modelling.

5.6 Cement and stray materials

Cement and other construction materials are being used in the development of ONKALO and will be used in the construction and operation of a repository, whether based on the KBS-3V or on the KBS-3H alternative. Detailed descriptions and inventories of cement and other construction materials are presented in /Hagros 2007a and 2007b/.

Cement will be used in the drift end plug and could also be used, in smaller amounts, in compartment plugs and to fill e.g. anchoring holes or grooves or as a grouting material. In the deposition drifts, it is currently foreseen that low-pH cement and/or Silica Sol (colloidal silica) grouts[15] will be used. These may have reduced detrimental effects on other system components (notably the buffer) compared with Ordinary Portland Cement (OPC) because of the reduced alkalinity of their porewater leachates, although the long-term durability and evolution of such materials is not as well understood as for the OPC.

[15] Silica Sol (colloidal silica) is being considered as a grouting material for narrow fractures /Autio et al. 2007/. Colloidal silica is a stable dispersion of discrete nonporous particles of amorphous silicon dioxide (SiO_2). The long-term durability and evolution of colloidal silica grouts has not been as well characterised as that of cementitious grouts /Ahokas et al. 2006/. For example, the potential for forming colloids with radionuclides is under investigation.

Where practicable, the introduction of potentially detrimental stray materials into the repository will be avoided, or such materials will be removed or cleaned during repository operations. Some stray materials will, however, inevitably remain in the repository and the surrounding rock after the closure. Potential amounts have been estimated, based on the current reference layout /Hagros 2007a/.

Stray materials of interest to safety assessment include those containing organic substances or nitrogen compounds, such as ammonium nitrates and NOx species injected into bedrock during blasting, which could have detrimental effects e.g. on the stress corrosion of copper. These are, however, generally expected to be decomposed or reacted relatively soon after blasting and it is likely that there will be limited amounts of these present after the operational period. Organic substances or their degradation products could, however, form complexes with radionuclides that would lower radionuclide sorption and thus increase radionuclide release and transport rates in the event of canister failure. These organic substances and their association with some metals could be particularly relevant when assessing the chemical risks associated with the repository system. The impact of these substances on the radiological consequences of the repository remains an issue for future safety assessments.

6 Overview of processes and evolution in successive time frames

6.1 Overall evolution

The timescales for the overall evolution of the repository and the site are illustrated in Figure 6-1. The figure assumes a repetition of the last glacial cycle, although the future evolution of the climate and the occurrence of future glaciations are subject to considerable uncertainty, particularly in regard to the long-term impact of anthropogenic emissions, especially greenhouse gases. The figure is adapted from a version developed for a KBS-3V repository evolution in /Pastina and Hellä 2006/. The timescales shown are the same in the KBS-3H case.

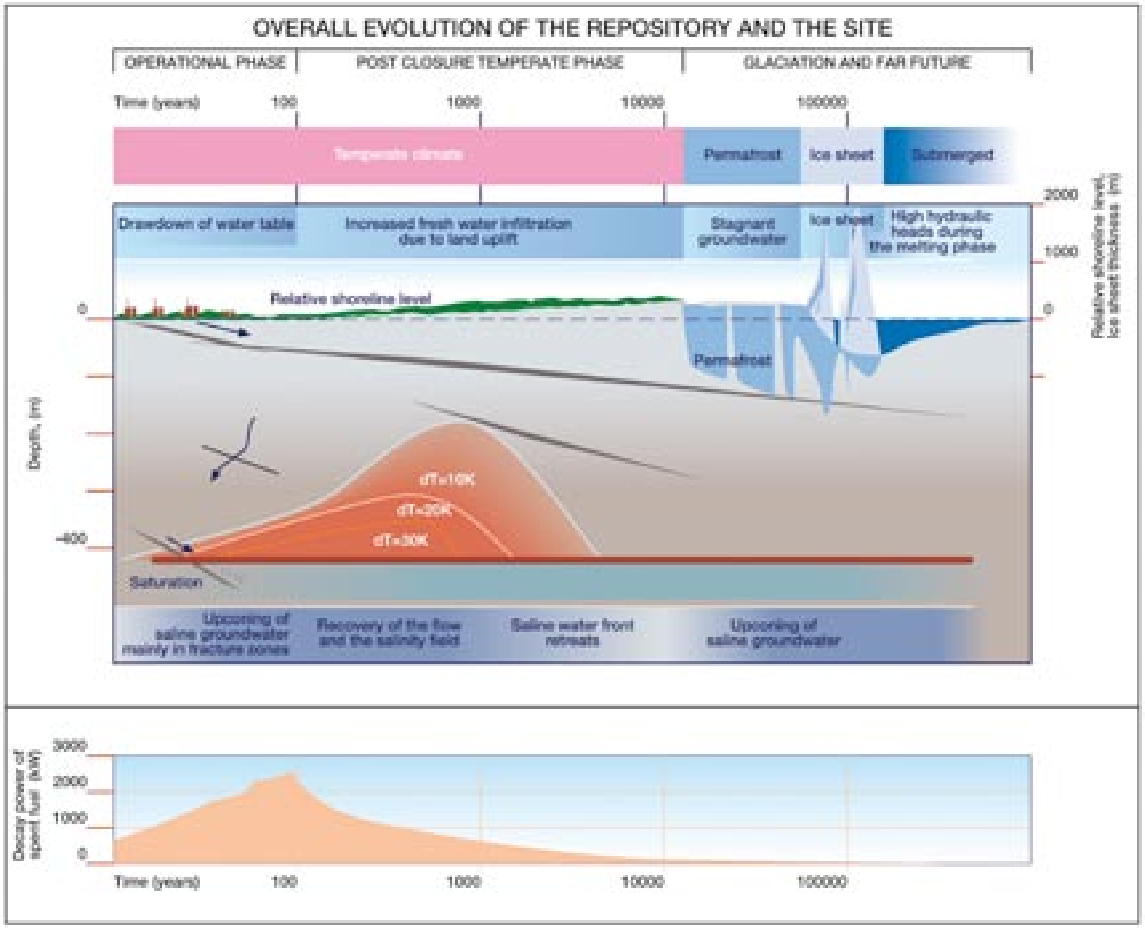


Figure 6-1. *The timescales for the overall evolution of the repository and the site under the assumption of a repetition of the last glacial cycle. The blue arrows show groundwater flow, black lines and dark grey areas mark fractures and deformation zones.*

6.2 Early evolution

6.2.1 General description

Early in the evolution of the repository, mass and energy fluxes will occur as a result of the various gradients created by repository construction and emplacement of the spent fuel. In both the KBS-3H and KBS-3V repository alternatives, the system evolves from its initial state through an early, transient phase towards a quasi-steady state, in which key safety-relevant physical and chemical characteristics (e.g. temperature, buffer density and swelling pressure) are subject to much slower changes than in the transient phase. It is in the early, transient phase that most of the significant differences in evolution between the KBS-3H and KBS-3V respository alternatives arise, although there are also some differences at later times. For example, the radionuclide transport paths from a failed canister are affected by the differences in the geometry and backfilling of the KBS-3H deposition drifts compared with the KBS-3V deposition tunnels. It is, however, the early, transient evolution of the repository and, in particular, the evolution of the buffer to a state of full saturation that has received particular attention in the KBS-3H safety studies.

Early evolution is taken to start with the emplacement of the first canister in the repository. The end-point is not well defined; many of the transient processes that occur during this period do not suddenly cease, but rather gradually diminish over time. Nevertheless, two key transient processes – heat dissipation from the spent fuel and saturation of the repository external to the canisters – may take up to several thousands of years, and even longer in the case of saturation of the tightest sections, and this may be taken as the rough duration of the early evolution period.

External conditions are expected to vary substantially over this period, with significant uncertainties associated with the impact of anthropogenic emissions, especially greenhouse gases. Although the global average sea level is expected to rise due to anthropogenic effects, continuing land uplift means that the shoreline is nevertheless likely to be displaced away from the Olkiluoto site. Within a few thousand years, the sea will have no impact on groundwater flow and composition around the repository, and thus the impact of anthropogenic changes in global average sea level on repository evolution will be negligible. Figure 6-2, shows a predicted evolution of the shoreline, the surface terrain and surface water bodies over the next seven thousand years.

The approximate timescales of key near-field processes and changes in key characteristics of the near field that take place during the early evolution of the repository are illustrated in Figure 6-3 (note that changes occurring within this period affect the characteristics and performance of the repository at later times, which is why the time-axis in Figure 6-3 extends to 10^6 years).

The heterogeneity of the host rock and the consequent spatial variability of groundwater inflow along the deposition drifts affect the timescales of several important processes. Figure 6-3 therefore distinguishes between processes with timescales that apply (i), at all locations within the drifts, irrespective of local groundwater inflow, (ii), in "less tight" drift sections, defined as drift sections in which the average hydraulic conductivity of the adjoining wallrock is about 10^{-12} m s^{-1} or more, (iii), in "tighter" drift sections, defined as drift sections in which the average hydraulic conductivity of the rock is in the range 10^{-13} to 10^{-12} m s^{-1}, and (iv), in "tightest" drift sections, defined as drift sections in which the average hydraulic conductivity of the rock is in less than 10^{-13} m s^{-1}.

It may not, in practice, be possible to differentiate tightest drift sections from tighter drift sections, as defined above, since neither may give rise to detectable initial water inflow during repository construction and operation. Nevertheless, the existence of these tightest drift sections cannot currently be excluded, and certain processes, such as saturation of the buffer may proceed somewhat differently compared with other drift sections, due to their tightness with respect to both groundwater flow and the flow of repository-generated gas.

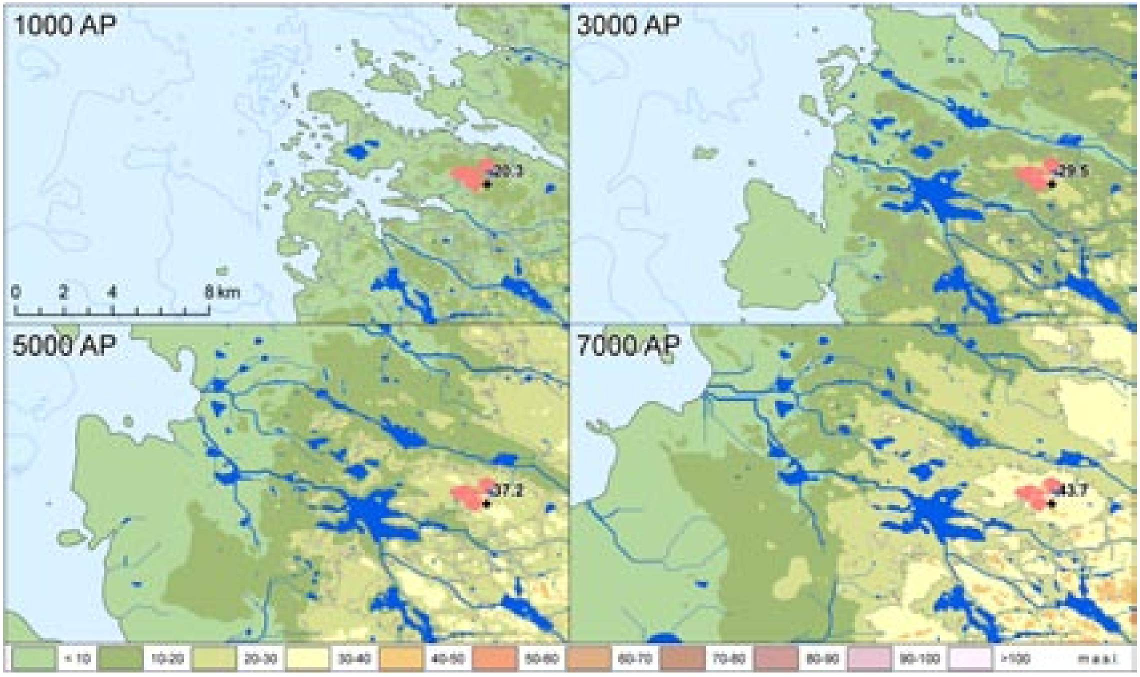


Figure 6-2. *Terrain and surface water body forecasts for selected times, assuming a repetition of the last glacial cycle. Colours show elevation in intervals of 10 metres. The present coastline is shown with a grey line, 10-m depth contours of the sea in blue, and illustrative repository layout in red /from Ikonen et al. 2007/. Top elevation of the hill above the ramp entrance is indicated in for each time point (m.a.s.l.).*

6.2.2 Effects of repository excavation on groundwater flow and composition

As noted in Section 5.1.7, excavation of ONKALO and of the repository at Olkiluoto may cause a transient drawdown of the water table /Pastina and Hellä 2006/ and an associated reduction of the hydrostatic groundwater pressure at repository depth. This could lead to the drawdown of fresh and brackish, sulphate-rich water from closer to the surface and upconing of more saline, deep groundwaters in the period up to saturation of the repository. However, measures to avoid these hydrological disturbances are being taken during the construction of ONKALO and will also be implemented during repository construction. So far, the observed hydrological disturbances have been minimal and this is expected to remain the case in the future /e.g. Ahokas et al. 2006/. The hydrostatic groundwater pressure will recover within a couple of years of backfilling and sealing the facility, although perturbed conditions may persist in the immediate vicinity of the repository due to the residual thermal output from the spent fuel, which gives rise to upward convective flow of deep, saline groundwater and increased salinity at repository depth compared with unperturbed conditions lasting for a period of up to several hundreds of years.

6.2.3 Thermal evolution

Heat generated by the radioactive decay of the spent fuel inside the canisters will be transferred continuously to the surrounding media by conduction, radiation from surfaces, and convection of gas or water in gaps and voids in the deposition drifts. Model calculations accounting for the first two of these transfer mechanisms are reported in /Ikonen 2003/ and /Ikonen 2005/. Results show that, for a canister pitch of 11 m and a drift separation of 25 m, temperature at the canister/bentonite interface at the centre of the highest-temperature canister peaks at 90°C after 50 years of repository operation – or about 20 years after deposition of the canister (Finnish BWR spent fuel). For a drift spacing of 40 metres and the same canister pitch, the temperature maximum is

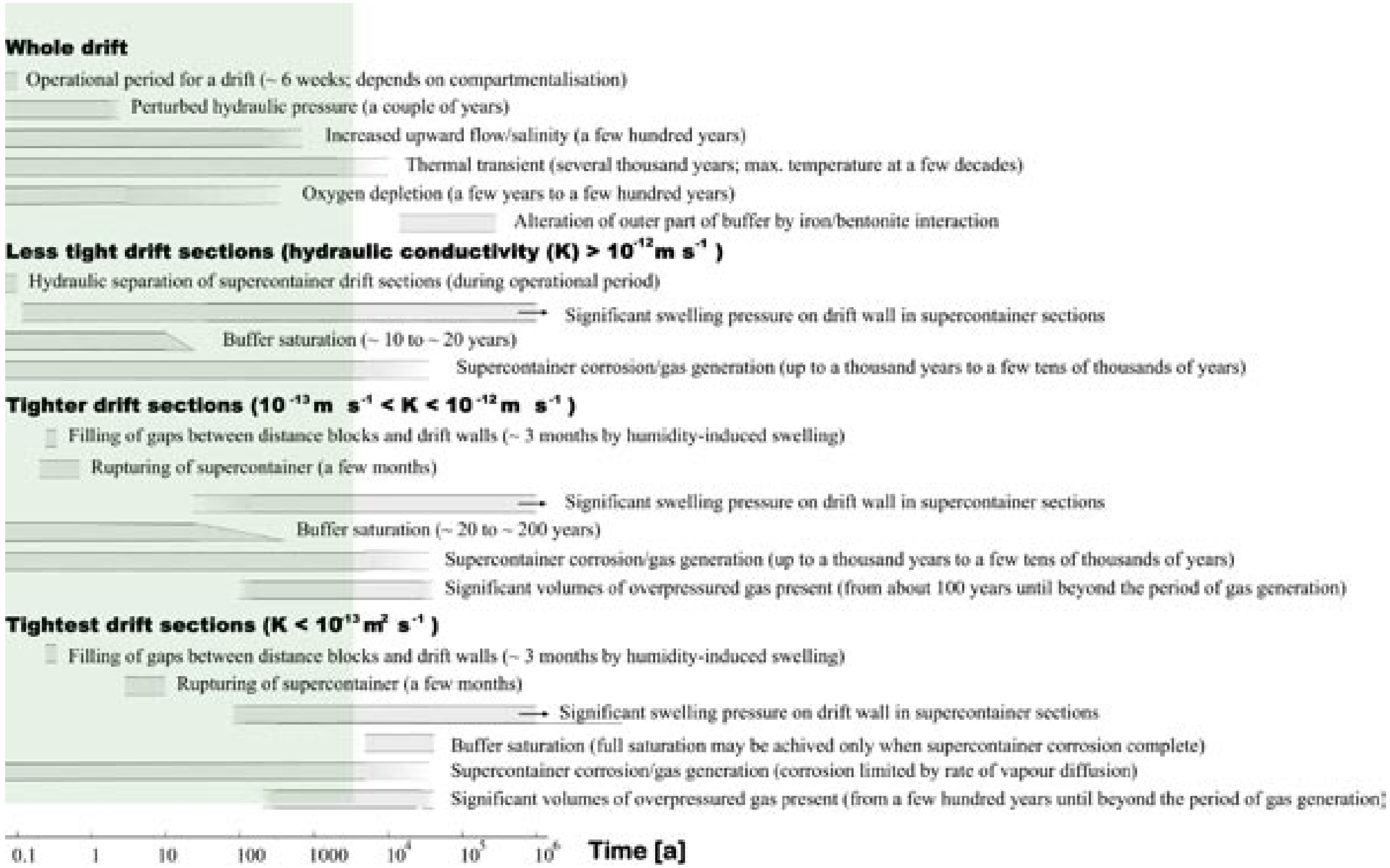


Figure 6-3. *Approximate timescales of various aspects of system evolution for a KBS-3H repository at Olkiluoto. Uncertainties are indicated by dotted lines. Tapering of the bars is used to indicate spatial variability of process timescales along the length of a deposition drift (e.g. as a result of geological heterogeneity). Background shading indicates the approximate duration of the transient phase (although some of the slowest transient processes may extend beyond this period). Right arrows indicate expected continuation beyond the million-year period covered by figure.*

slightly lower (84°C) and reached somewhat earlier (after 40 years of years of repository operation – or about 10 years after deposition of the canister). During the period of elevated temperature, some limited chemical changes may occur in the buffer (Section 6.2.5). Furthermore, some thermally-induced rock spalling is expected in those drift sections where limited groundwater inflow means that significant buffer swelling pressure on the drift wall takes a few years or longer times to develop. Thermally-induced rock spalling is judged to be a potentially important safety issue, the consequences of which are discussed in Section 7.1.6.

The decay heat of spent nuclear fuel will continue to raise the temperature of the repository and the surrounding bedrock by several tens of degrees for many centuries and by a few degrees for several thousand years.

6.2.4 Buffer saturation, gas generation and the development of swelling pressure

The distance blocks and the buffer inside the supercontainers will take up water flowing into the drift through intersecting transmissive fractures and water vapour in the air. Water vapour in the air in the deposition drifts will originate from the rock and, during the operational period, from the water cushion system that is proposed for the deposition vehicle /Autio et al. 2007/.

Water uptake will result in swelling of the buffer, and the establishment of a tight contact between the distance blocks and the drift wall. In drift sections where water inflow occurs through more transmissive fractures ("less tight" drift sections, with wall rock hydraulic conductivities, K, of more than 10^{-12} m s^{-1}), hydraulic separation between the supercontainers is likely to develop during the operational period. In relatively tight drift sections, where the wall rock hydraulic conductivity is less than about 10^{-12} m s^{-1}, humidity-induced swelling will require about three months to provide hydraulic separation between the supercontainers. In the case of the DAWE design variant referred to in Section 4.1, drainage will prevent significant water uptake and swelling during the operational period. Following artificial watering at the end of operations, however, hydraulic separation between the supercontainers will be rapidly achieved.

Uptake of liquid water by the buffer will lead to some extrusion of low-density bentonite through the perforations in the supercontainer shell. As water is taken up, swelling pressure differences between the inside and outside of the supercontainer shell will cause the supercontainer shells to deform and possibly rupture (see Section 5.4.2 of the Evolution Report, /Smith et al. 2007b/). The various strain mechanisms that are involved in the early evolution of the supercontainer shells could have a detrimental effect on the outer part of the buffer depending on how the shells deform and they may generate heterogeneities in the buffer. This is another issue for further study (Section 11.2.2). Especially in relatively tight drift sections, the possibility that the shell could subsequently peel open (probably along its weld) and partially contact the rock cannot currently be excluded. This could occur within a few months of emplacement. Contact between the steel supercontainer shell (and its subsequent corrosion products) and the drift wall is judged to be a potentially important safety issue, the consequences of which are discussed in Section 7.1.3. In wetter conditions, the supercontainer will also rupture, but extruded bentonite already present outside the supercontainer before the swelling pressure becomes large enough to cause rupturing is likely to prevent contact with the drift wall.

Heterogeneity of the near-field rock results in widely differing saturation times for different drift sections. A significant swelling pressure on the drift wall adjacent to the supercontainers is likely to develop in no more than a few years in drift sections where the wall rock hydraulic conductivity is more than about 10^{-12} m s^{-1}, while full saturation of the buffer will take about a decade. In such sections, the time taken for the buffer to saturate will be controlled by the ability of the bentonite to take up water. In drift sections drift sections with hydraulic conductivities in the range of about 10^{-13} m $s^{-1} < K < 10^{-12}$ m s^{-1}, the rate of saturation is limited by the tightness of the rock. In drift sections where $K < 10^{-13}$ m s^{-1}, the high gas transport resistance of the rock is likely to mean that hydrogen gas from the corrosion of steel components external to the canister accumulates at high pressure in the void space around the supercontainer. An equilibrium situation will be reached between gas generation and the outward migration of gas by:

- transport along the deposition drift via the distance block/rock interface or through the EDZ, eventually reaching either a drift section in fractured rock or the transport tunnel system, or
- reactivation of existing but hydraulically tight fractures.

Once the equilibrium gas pressure increases above hydrostatic pressure (i.e. overpressures are developed), this will further limit or prevent groundwater inflow from the rock and delay buffer saturation.

Calculations reported in Appendix C of the Process Report indicate, however, that gas pressures will be insufficient to reactivate hydraulically tight fractures or to form pathways along the bentonite/host rock interface, and that the EDZ is a potential migration route for gas away from tight drift sections. These calculations are, however, based on assumed EDZ properties that are uncertain. The better characterisation of the EDZ is an issue for further studies (Section 11.2.4).

The potential effects of gas from steel corrosion on buffer saturation and canister corrosion are judged to be further important safety issues, and are discussed in more detail in Section 7.1.4. Gas overpressures in the tightest drift sections may persist until the supercontainers and other steel components are fully corroded. This could take up to several thousands of years based on an expected steel corrosion rate of 1 $\mu m\ a^{-1}$. Even in the tightest drift sections, however, the buffer is expected to retain its initial water content, and will eventually fully saturate, at which time it is expected to perform its full range of safety functions.

Because of the heterogeneous rate of saturation of the drift, significant transient pressure differences may develop between water in the saturated gaps in less tight drift sections, which will be close to hydrostatic pressure, and other more slowly saturated drift sections where pressures are lower. If high pressure differences develop too rapidly, before sufficient swelling pressure develops at the interface between the distance blocks and the drift wall, this could result in transient water flows („piping“) along the interface, which could in turn lead to erosion and loss of buffer mass in some supercontainer sections, and possibly an increase in mass in others. A loss of swelling pressure due to piping and erosion could also lead to enhanced spalling due to reduction in confining pressure associated with time-dependent degradation of rock strength. Thermally-induced rock spalling is judged to be a potentially important safety issue, the consequences of which are discussed in Section 7.1.6, and it also remains an issue for further study (section 11.2.3). Design measures will be taken to reduce the possibility of significant piping and erosion during early evolution of the buffer (see the discussion of groundwater control and compartmentalisation of the drifts in Section 4.2). Currently, however, the possibility of some degree of piping and erosion during buffer saturation cannot be completely eliminated, and thus the issue of piping and erosion remains a potentially important safety issue, the consequences of which are discussed in Section 7.1.2.

The possibility of canister sinking through the buffer is discussed in Section 3.6.1 of the Process Report, Section 6.4.3 of the Evolution Report (and references therein). In Table 4-2, a safety function indicator criterion for the minimum swelling pressure to prevent significant canister sinking has been set to 0.2 MPa. The criterion was taken over from the SR-Can safety assessment of a KBS-3V repository /SKB 2006a/. For a given buffer density, any canister sinking that occurs is expected to be less for KBS-3H than for KBS-3V, due to the fact that the weight of the canister is distributed over a greater area than in KBS-3V. The 0.2 MPa criterion is therefore expected to be conservative in the case of KBS-3H. The scoping calculations described in Section 4.6.1 of the Process Report and Appendix B of the Evolution Report indicate that none of the processes occurring during that early evolution of the repository that are of particular relevance to KBS-3H (e.g. axial displacement of distance blocks and supercontainer, piping/erosion) have the potential to cause a decrease of buffer density below the safety function indicator criterion. Canister sinking is, therefore, not further discussed in the present report.

6.2.5 Chemical, mechanical and microbiological processes in the near field

As the buffer saturates, it will evolve chemically, important processes being ion exchange and the dissolution/precipitation of calcite. Much of the oxygen trapped in the drift will migrate by diffusion to the buffer/rock interface, where it will be consumed over a period of a few years to a few decades principally by fast microbially induced reactions, and also by inorganic reactions with the KBS-3H steel supercontainer and other steel components. Within the buffer itself, the development of a fine, homogeneous pore structure and high swelling pressure means that microbes are not expected to stay active and viable.

Cementation by precipitation of sulphates due to elevated temperatures around the canisters may lead to the formation of a zone in the buffer adjacent to the canisters with increased strength but reduced plasticity. As temperatures and temperature gradients decrease however, it is likely that the precipitated solids will redissolve, and disperse within the buffer by diffusion.

Silica in the buffer close to the canister is expected to dissolve during the period of elevated temperature and be transported outwards by diffusion to colder parts where precipitation may take place. Buffer cementation could in principle take place due to the dissolution, transport and precipitation of silica or aluminosilicate minerals, but neither experimental nor natural analogue studies have shown that this process will actually occur. The effect of buffer cementation due to silica precipitation is, however, an issue for further work (see Section 11.2.2).

Strongly reducing conditions will prevail in the buffer following the depletion of trapped oxygen. Under such conditions, structural iron in smectites, which occurs in the buffer mainly as Fe(III), may be reduced to Fe(II) by electron transfer between the species in solution and in the structure. Reduction of Fe(III) to Fe(II) could, in principle, have consequences for the stability and to some extent the swelling pressure and hydraulic properties of the buffer. There are indications from experiments that the effects may be small, although they remain an issue for further study (Section 11.2.3).

Radiolytic effects outside the canister can be neglected because of the initially low gamma dose rate at the canister surface (less than 1 Gy per hour by design) and because gamma radiation, mainly due to Cs-137 decay decreases by a factor of 1,000 within 300 years /SKB 2006a, Section 6.4.1/. Alpha and beta radiation are shielded by the copper canister and therefore do not cause any radiolysis outside the canister.

During the unsaturated phase, nitric acid formed by gamma radiolysis of residual moist air around the canister will contribute to the corrosion attack of copper during the early evolution. As shown in SR-Can's fuel and canister process report, the amount of nitric acid formed corresponds to a corrosion depth of only a few microns /SKB 2006d, Section 3.5.4/.

As mentioned above, steel components external to the canister will corrode. Iron corrosion products have a low solubility under expected repository conditions, but will nevertheless slowly dissolve, releasing Fe(II) to the buffer porewater. The subsequent interaction of Fe(II) with bentonite, which could potentially include mineral transformation of smectite to a non-swelling clay, is the subject of ongoing research. Mineral transformation of the buffer is, however, expected to proceed slowly, due to the slow diffusive migration rate of Fe(II), which is retarded by sorption, and the slow kinetics of the transformation processes. To estimate the timescales required for transformation of a significant proportion of the buffer, a 1-D reactive transport model was set up using site-specific information from Olkiluoto, such as groundwater and mineralogical data /Wersin et al. 2007/. Preliminary modelling results indicate that the extent of the zone potentially undergoing mineral transformation is likely to remain spatially limited (a few centimetres) for very long times (hundreds of thousands of years or more). A key safety issue is considered to be the impact on mass transfer at the buffer/rock interface, as discussed in Section 7.1.3. Another aspect is the potential cementation effects by mineral precipitation resulting from Fe-bentonite interaction. Available data is by no means conclusive on this issue. Examples from natural systems reported refer to cementation effects induced by iron oxide precipitation under oxic, but not anoxic conditions. There is no natural analogue known to us which would indicate significant cementation effects in swelling clays under anoxic conditions. The cementation effect as a consequence of Fe-bentonite interaction is still not well understood and is an issue for further investigation (section 11.2.3).

Some interactions involving the limited amount of low-pH cement or Silica Sol and stray materials left in the repository may also occur within the buffer and host rock, a key safety issue again being the impact on mass transfer at the buffer/rock interface, as discussed in Section 7.1.5.

Hydrogen gas from the corrosion of steel components may create transport pathways through the buffer, but buffer swelling pressure is expected to reseal these pathways once the gas generation ceases and gas pressure falls. Along with methane and hydrogen naturally present in the groundwater, it may also participate in the reduction of groundwater sulphate to sulphide in the presence of sulphate-reducing bacteria, increasing the sulphide concentration and the rate of canister corrosion. The impact of hydrogen gas on the bentonite porewater chemistry, e.g. on the acid-base equilibria, has not been fully evaluated, and is noted as an issue for further study in Section 11.2.6.

During early evolution, shear movement on fractures intersecting the deposition drifts may take place as a result of rock excavation and heat load, leading to mechanical forces on the canisters. The plasticity of the buffer is, however, expected to protect the canisters from such movements. Based on the lack of indications of recent seismic activity at Olkiluoto (Section 5.1.2), the probability of a tectonic earthquake large enough to cause damage to the canisters is expected to be small during this period.

6.2.6 Evolution of the canister

As the buffer saturates and swells, it will exert an increasing mechanical load on the canisters. Non-uniform buffer wetting and build-up of swelling pressures causing bending of the canisters in the supercontainers is likely in the very early stage. However, this non-uniform swelling does not cause cracking of the copper overpack, nor of the weld, as shown in Posiva's design report /Raiko 2005/ and SKB's weld test results reported in the FUD programme description /SKB 2007/.

Posiva's canister design report /Raiko 2005/ contains a number of conservatively selected load cases that describe the load conditions caused by the possibly non-uniform wetting. The iron insert is shown to be strong enough to withstand the postulated loads. The swelling pressure causes displacement-controlled load that, in turn, causes secondary stresses. Secondary stresses are not very harmful, because, even in case of yielding, the strains do not increasing remarkably. The important characteristic of secondary stresses is that the load decreases and eventually vanishes when the structure is deformed due to elastic and plastic deformation.

According to the strength analyses in the canister design report /Raiko 2005/, the strains in the cast iron insert are usually elastic and only in the worst case slightly plastic, but in no case the strains are high. The elastic modulus of copper is only half of that of iron. Thus the stresses in the copper liner around the iron insert are only about half the stresses in iron. The plastic ductility of copper is much higher than that of iron. Thus there is no risk for copper cracking or tearing in the kind of postulated load condition.

As for the voids in copper welds acting as crack initiation nuclei in yielding or creeping condition, the actual creep tests made with actual copper weld material (both FSW- and EB-welds including typical micro-faults) have not yielded to cracking or tearing in low strain level, but only with high strain level (20–30% for EB-welds and > 40% for FSW-weld) according to /SKB 2007/. Any differences in density and swelling pressure around individual canisters and also along the drift are expected to diminish over time due to homogenisation of the bentonite and, although some inhomogeneity may remain, the load on the canisters is expected to become approximately isostatic (i.e. equally large over the entire canister surface area) and similar for all canisters. The isostatic load on the canisters following buffer saturation will be about 11–12 MPa.

The corrosion of copper in the first several thousand years after emplacement is discussed in detail in Sections 6.1.4 and 7.1.4 of /Pastina and Hellä 2006/ in the case of a KBS-3V repository at Olkiluoto, and is likely to be similar in the case of KBS-3H. The presence of initially entrapped oxygen will cause some limited corrosion of copper until the available oxygen is depleted and the environment changes from oxic to anoxic. Thereafter, the only significant corrosion agent will be sulphide that diffuses slowly through the buffer to the canister surface.

Scoping calculations of canister lifetime as a function of sulphide concentration of the groundwater for a range of cases and variants are presented in Appendix B.7 of the Evolution Report, assuming the canister to be located in a drift section intersected by a fracture with a transmissivity towards the upper end of the expected range. The variants illustrate the impact on canister lifetime of features and processes with the potential to increase the rate of transport of sulphide across the buffer/rock interface and across the bulk of the buffer, including:

- piping and erosion,
- thermally-induced rock spalling,
- the presence of potentially porous or fractured corrosion products in contact with the drift wall in relatively tight drift sections with wall rock hydraulic conductivity less than about 10^{-12} m s^{-1},
- chemical interactions of the buffer with these corrosion products,
- chemical interactions of the buffer with high-pH leachates from cementitious components.

The results of these calculations are shown in Figure 6-4.

Case (a) corresponds to the situation where the buffer/rock interface is unperturbed. Two variants are considered in this case: (i), in which diffusion in the buffer takes place under the assumption that the bulk of the buffer is also unperturbed, and the rate of diffusion of sulphide is evaluated assuming that the density of the saturated buffer reaches its design value of 2,000 kg m^{-3}, and (ii), in which the rate of diffusion in the buffer is assumed to be increased due to limited density loss, e.g. because of piping and erosion. Cases (b) and (c) correspond to situations where the buffer/rock interface is perturbed by one or more of the other processes listed above. The affected zone is treated as a highly conductive "mixing tank". In case (b), the mixing tank is confined to a narrow region around the outer edge of the buffer. Two variants are considered: (i), in which the mixing tank has negligible thickness, and (ii), in which it occupies 10% of the buffer thickness. Finally, in Case (c), the mixing tank is assumed to extend throughout the buffer.

The calculated corrosion rate is sensitive to the groundwater flow rate at the buffer/rock interface, which is expected to be highly variable along the drift. The model used for the scoping calculations assumes that the drift is intersected near the canister location by a fracture with a transmissivity of 3×10^{-9} m^2 s^{-1}. As discussed in Appendix B of the Evolution Report, this is a moderately pessimistic assumption, since the transmissivity is at the higher end of the expected range on the basis of the current 0.1 litres per minute inflow criterion for drift sections that would be accepted for canister and buffer emplacement (Section 4.2). However, as noted in the Evolution Report, the possibility of some higher transmissivity fractures intersecting the drift near a canister location cannot be excluded.

With no perturbation to the buffer/rock interface, a canister lifetime in the order of millions of years is calculated irrespective of the assumed buffer diffusion coefficient if the highest measured sulphide concentration at repository depth at the site is assumed. It is acknowledged, however, that there are several simplifying assumptions in the scoping calculations that could result in somewhat higher or lower canister actual lifetimes than those calculated. In particular, while sulphide is abundant in the groundwater at repository depth (Section 5.1.4), there is some uncertainty in the evolution of sulphide concentration over time.

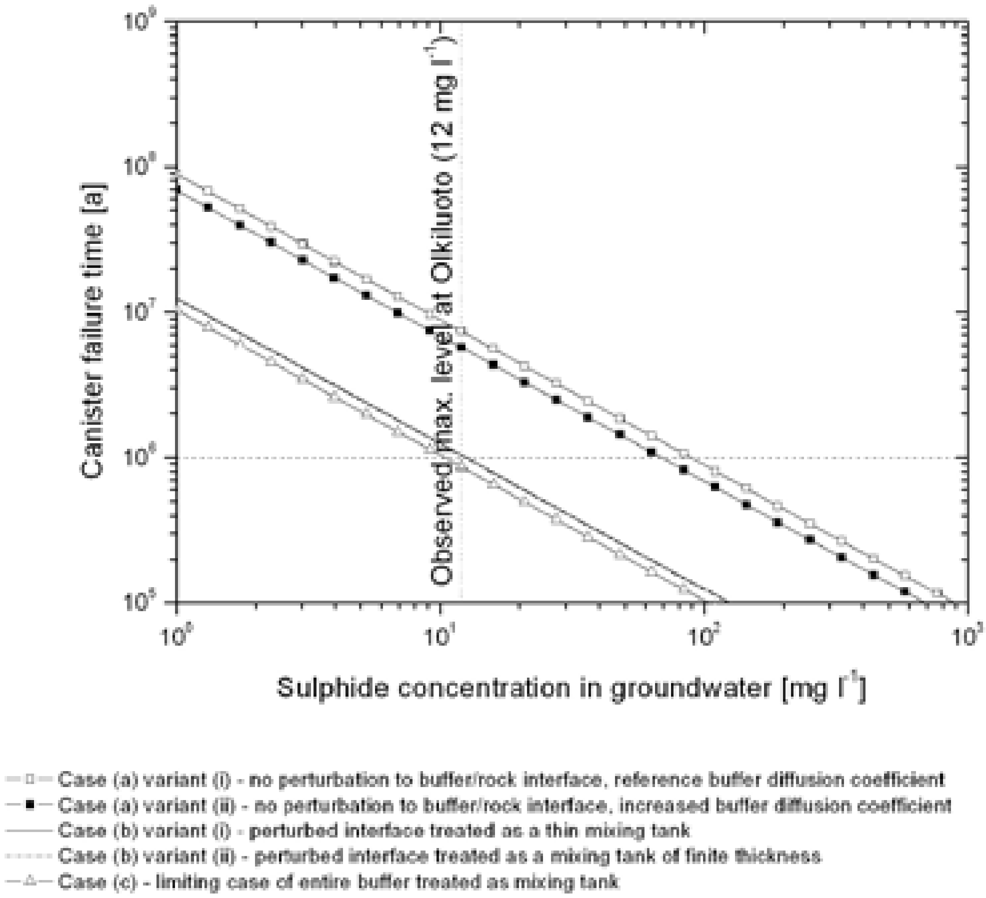


Figure 6-4. *Canister failure time as a function of groundwater sulphide concentration for the cases defined in the main text. Note that the results for case (b), variants (i) and (ii), approximately coincide. See Appendix B.7 of the Evolution Report /Smith et al. 2007b/ for details.*

As noted in Chapter 11, the impact of dissolved methane and hydrogen on sulphide concentration and canister corrosion is an issue for both the KBS-3H and KBS-3V repository alternatives that required further investigation. The methane and hydrogen naturally present in the groundwater could, along with the hydrogen generated by the corrosion of the steel repository components external to the canister[16], participate in the reduction of groundwater sulphate to sulphide in the presence of sulphate-reducing bacteria, increasing the sulphide concentration.

The supercontainer and other steel components also have the favourable effect (not included in the scoping calculations) that some of the sulphide entering the buffer will react with their corrosion products, and with amorphous iron present in the bentonite. The resulting iron sulphides have a low solubility and will precipitate, reducing the flux of sulphide to the canister surface. The supercontainer will add as a sink of the generated sulphides, which may consequently increase the corrosion of the supercontainer.

The presence of a thin mixing tank and the resulting higher flux of sulphide to the canister surface give a significantly reduced canister lifetime. Canister lifetime is further reduced, though only slightly, if the mixing tank extends to a finite depth into the buffer (10% of the buffer thickness). The impact of treating the entire buffer thickness as a mixing tank is also quite limited, indicating that the transport resistance provided by the buffer does not dominate the transport of sulphide to the canister surface. Even in this extreme case, the canister lifetime falls below a million years only if sulphide concentrations above the currently observed maximum level at

[16] The impact of repository-generated hydrogen on canister lifetime is, however, expected to be small, since hydrogen is generated over a limited period of time – see Section 7.1.4.

Olkiluoto are assumed. Only in the case of a high sulphide concentration of 100 mg per litre being maintained over a prolonged period at the buffer/rock interface, e.g. by microbial activity, is canister lifetime reduced below about 100,000 years (Figure 6-4).

Overall, although some uncertainties remain regarding the rate of corrosion, the copper coverage of the canisters is expected to remain intact during early evolution, and with a large safety margin, assuming appropriate quality control procedures during canister encapsulation and emplacement, such that there are no initial penetrating defects. As a consequence, spent fuel and radionuclides are expected to be totally isolated within the canisters throughout this period.

6.3 Subsequent evolution prior to future major climate change

Following the period of early evolution, the repository and its geological environment will evolve to a quasi-steady state, in which key safety-relevant physical and chemical characteristics are subject to relatively slow processes.

By definition, the heat output of fuel will have declined to a level that has no significant effect on the evolution of the repository and its environment. Minor variations in climate prior to any future glacial episode are also not expected to have any significant impact on temperatures at repository depth. The reduction in thermal output from the fuel will result in rock stress levels returning to those prior to spent fuel emplacement and in a reduction in the thermo-mechanical loads on the canister.

The buffer will continue to evolve chemically. Mineral transformation of the buffer due to the presence of Fe(II) from corroding steel components is also likely to extend further into the buffer. As noted in Section 6.2.5, however, the results of the preliminary 1-D reactive transport modelling reported in /Wersin et al. 2007/ indicate that the Fe(II) front migrating inwards from the supercontainer will penetrate only a few centimetres into the buffer, even after hundreds of thousands of years.

As a result of continuing land uplift prior to any future glaciation, the saline water currently at repository depth will start to be replaced by brackish, sulphate-rich and fresher groundwater currently present at shallower depths, and the driving forces for groundwater flow will also change. The changes in the salinity of the groundwater will affect the swelling pressure of the buffer, with lower salinities giving higher swelling pressures.

The copper canisters will continue to slowly corrode. Given the expected anoxic conditions at the canister surface, the only corrosion agent will be sulphide that diffuses through the buffer from the rock. Because of the small diffusive flux of sulphide that is expected to reach the canister surface, the rate of copper corrosion will be very low. Although there is some uncertainty related to perturbations to the buffer/rock interface (e.g. due to thermally-induced rock spalling, the presence of iron corrosion products from the supercontainer shell and iron/bentonite interaction), scoping calculations described in Appendix B.7 of the Evolution Report indicate that a combination of pessimistic assumptions must be made before the canister lifetime drops to a million years or less. No canister failure by corrosion of the copper shell is expected prior to the next glacial episode.

There is likely to be some intrusion of buffer material into open fractures intersecting the deposition drifts. A sufficient reduction in the salinity of the groundwater over time could make the extruded buffer material more susceptible to erosion by flowing groundwater. This is a particular concern in the event of a prolonged period of temperate climate due to anthropogenic emissions, especially greenhouse gases, and hence a prolonged period of continuing land uplift. As discussed in Section 7.2.7, significant erosion of this material by flowing water is not expected, irrespective of any anthropogenically induced changes in climate, although further study would be beneficial to reinforce this conclusion.

6.4 Effects of major climate change

In general terms, the changes expected in the surface environment during a future glacial cycle are as follows:

- the glacial cycle will be initiated by a cooling of the climate,
- as the climate grows progressively cooler, periglacial conditions will become established; at first, the land surface temperature will fluctuate around freezing point, and then will fall further, such that the ground remains frozen year round (permafrost),
- at some point, an advancing ice sheet will reach the repository location, thicken gradually and then retreat relatively rapidly by melting,
- after the ice has gone, periglacial conditions will resume, characterised by permafrost and near-freezing average annual air temperatures,
- a warming trend will restore average annual temperatures to near-present-day values.

During future glaciations, the load of up to 2 km-thick ice sheets is expected to depress the Earth's crust, leaving the Olkiluoto area submerged below sea level. The site will experience a gradual uplift following ice-sheet melting and return to the temperate condition, as in past glacial cycles.

The development of permafrost and frozen ground will have only a minor influence on hydro-mechanical pressure conditions at repository depth, possibly leading to a more stagnant flow pattern. The formation of ice sheets will, however, result in large water pressures and hydraulic gradients in the subsurface, and the possibility that large volumes of low-salinity and possibly meltwater could be forced into deeper parts of the bedrock must be considered in the safety assessment. When an ice sheet is present, seismic activity, which is in any case low in the Olkiluoto region, will be further repressed. Post-glacial earthquakes may, however, occur following the retreat of the ice sheet, giving rise to stress changes in the rock that trigger shear movements on smaller-scale fractures that intersect the deposition drifts. Assuming a repetition of the last glaciation, the next glacial retreat will be in about 70,000 years time (see, e.g. Figure 7-1 in the Evolution Report), although there are significant uncertainties, particularly in regard to the long-term impact of anthropogenic emissions, especially greenhouse gases.

Potentially significant effects of a change to much colder climatic conditions for the evolution of repository, which are common to the KBS-3H and KBS-3V designs, are described in Section 7.2. These include the possibilities of:

- buffer freezing (Section 7.2.2),
- canister failure due to isostatic load (Section 7.2.3),
- oxygen penetration to repository depth (Section 7.2.4),
- shear movement on fractures intersecting the repository drifts in the event of a large earthquake (Section 7.2.5),
- loss of buffer from exposure to dilute glacial meltwater (Section 7.2.6)

6.5 Evolution of a canister with an initial penetrating defect

The possibility that an initial defect will penetrate a copper canister shell (or be sufficiently deep to have significant implications for the timing of failure due to copper corrosion) is not currently excluded. The likelihood of this particular canister failure mode is discussed in Section 8.4.

The behaviour of the water/vapour/gas system in a defective copper-iron canister is complex. Water entering a canister through an initial penetrating defect will react with and corrode the cast iron insert, resulting in the formation of iron corrosion products, the weakening of the insert, the generation of hydrogen gas, and the interaction of iron corrosion products with bentonite located in the vicinity of the defect.

As internal gas pressure within the canister increases, the driving force for liquid water ingress will be reduced and possibly eventually reversed. Furthermore, as the void space fills with corrosion products, the corrosion rate will drop due to the increasing transport resistance between the defect and the uncorroded surfaces.

The canister insert will continue to be weakened by progressive corrosion, although the rate at which this will occur is uncertain. Provided the initial defect where penetration occurs remains small, possibly becoming plugged with bentonite or corrosion products, corrosion of the external surface of the insert is likely to drop to a very low rate, being controlled by the slow migration of water through the defect and corrosion product pore space to the corroding surface or the slow dissolution of the corrosion products and diffusion of the dissolved iron to the defect and into the buffer (Figure 6-5a). On the other hand, the mechanical pressure exerted on the copper shell by the corrosion products could potentially expand the original defect, allowing corrosion of the insert to continue at an increased rate (unlimited water supply), at least over the part of its surface nearest to the defect (Figure 6-5b).

Eventually, the canister will collapse. It is likely that the collapse of a weakened insert will take place during or after a major glaciation, although earlier collapse cannot be excluded.

The corrosion products from the insert may react with swelling clay minerals in the buffer in the vicinity of the canister defect, although their effect will extend only very slowly into the buffer. Corrosion of the insert may also expand the copper shell and lead to a compaction of the buffer, increasing the pressure exerted by the buffer on the canister and rock. A conservative scoping calculation in Appendix B of the Evolution Report indicates that corrosion of the insert could lead to swelling pressures of around 10 MPa or greater, depending on the salinity of the groundwater. A modelling study by /Lönnqvist and Hökmark 2007/ found that, for a KBS-3H repository at Olkiluoto, a pressure on the drift wall of 10 MPa or more might open pre-existing horizontal fractures intersecting the drift at mid-height, although the effects are expected to be small, in terms of increase in fracture aperture and distance from the drift wall to which such effects extend, even at pressures of 20 to 25 MPa. Nevertheless, a scenario in which damage to the rock affects radionuclide transport from a canister with an initial penetrating defect is identified in Chapter 8.

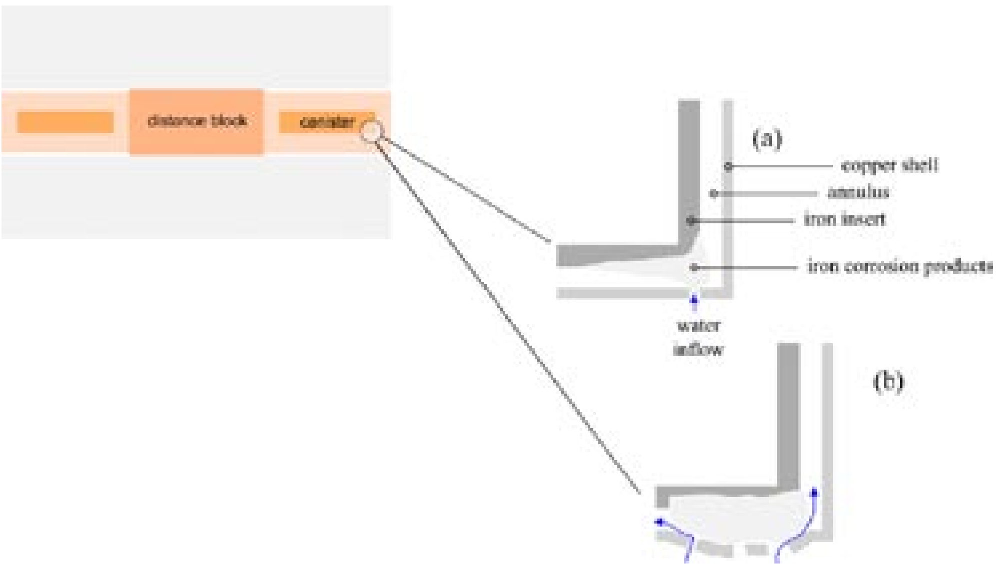


Figure 6-5. *Possible impacts of iron corrosion on water inflow – (a), corrosion products and bentonite plug the defect in the shell, providing a barrier to further inflow, (b), corrosion products expand the original defect, increasing water inflow.*

Once water penetrates the cast iron insert and enters the channels in the insert containing the fuel assemblies, it will corrode the channel surfaces, giving rise to further corrosion products and gas. It will contact the surfaces of the cladding and, if the cladding tubes are ruptured, it will also contact the surfaces of the fuel pellets and interact with them.

The cladding and other metal components will slowly corrode on contact with water, resulting in the congruent release to solution of activation products embedded within them. Once water contacts the fuel pellet surfaces, radionuclides that have been segregated to grain boundaries in the fuel, to pellet cracks and to the fuel/sheath gap will either rapidly enter solution, or, in the case, for example, of C-14, may form volatile products such as methane or carbon dioxide that will mix with hydrogen gas inside the canister. Water will also start to interact (but far more slowly) with the surfaces of the fuel matrix, and radionuclides embedded in the fuel matrix, along with radioactive gases present in fission gas bubbles, will be released as the matrix is slowly dissolved or otherwise altered.

Radionuclides released from the fuel (including activation products in the metal components) will either enter solution, form volatile species that can mix with repository-generated gas (particularly relevant for C-14), or, if their solubilities in water are low, precipitate either as immobile solids or as colloids. Dissolved radionuclides and radionuclides in gaseous form will be transported by gas and water movements (advection) and by diffusion in the interior of the canister, and may exit through the damaged copper shell. Radionuclides in colloidal form will also migrate in water in the canister interior by advection and diffusion, but will be prevented from migrating into the buffer by its microporous structure.

If the inflow rate of water into the canister exceeds the maximal consumption rate by steel corrosion, water will slowly accumulate within the cavity until the gas pressure is in equilibrium with the pressure outside the canister. Due to ongoing gas generation, the pressure inside the canister may further rise. As a consequence, any water previously accumulated within the canister will be slowly squeezed outwards through the hole, along with any radionuclides that have been released to solution in the intervening period. This possibility represents a key safety issue, and is discussed further in Section 7.1.7.

Induced fission (criticality) inside or outside a canister with an initial penetrating defect is not expected to occur. For example, for the three types of Finnish canisters, /Anttila 2005b/ has shown that the fuel remains subcritical even if the void inside the canister is entirely filled with water. This conclusion is expected to hold for both vertically and horizontally orientated canisters (see Section 8.11 of the Evolution Report).

7 Key safety issues and their consequences

7.1 Key safety issues in the early evolution of a KBS-3H repository

7.1.1 Overview of issues

The difference analysis of safety relevant features and processes in the KBS-3V and KBS-3H designs (Table 3-1), together with the description of system evolution presented in the KBS-3H Evolution Report and summarised in Chapter 6, indicates that there are key issues with the potential to significantly perturb the safety functions of the repository components that have a different significance to, or potential impact on, KBS-3H compared with KBS-3V. These issues have been the subject of KBS-3H-specific analyses. They concern mainly the early, transient evolution of the repository, although they may have implications for canister integrity and radionuclide release and transport in the longer term. They include:

1. piping and erosion during repository operations and drift saturation, which is discussed further in Section 7.1.2,
2. steel components external to the canisters, their corrosion products and their impact on mass transport – Section 7.1.3,
3. the effects of gas from the corrosion of these components – Section 7.1.4,
4. interactions involving leachates from cementitious components – Section 7.1.5,
5. thermally-induced rock spalling – Section 7.1.6,
6. expulsion of water and dissolved radionuclides from a defective canister interior by gas – Section 7.1.7.

The issues, associated uncertainties and the evaluation of impact of these on the mode and timing of canister failure and on subsequent radionuclide release and transport are summarised in Table 7-1. It should be noted that the most significant impact of issues 2, 4 and 5 is, in the first place, on the mass transport properties of the buffer and the buffer/rock interface (the other issues and associated processes may also have some limited effects on these properties). As a consequence, they may also affect canister lifetime by increasing the rate at which sulphide from the groundwater migrates to, and corrodes, the canister surface, as well as the rate at which released radionuclides are transferred across the interface.

Issues 1, 4 and 5 are issues of concern for both KBS-3H and KBS-3V, but are included here because their likelihood, extent or impact may be significantly different for the two alternatives. Piping and erosion could, for example, potentially occur within KBS-3V deposition holes. These can, however, be assessed individually, such that those with water inflows that could lead to piping and erosion are rejected.

It should also be noted that some of the differences identified in Table 3-1 have been found, on the basis of scoping calculations and qualitative arguments, not to be key safety issues, and are not included in Table 7-1 or discussed in the following sections of this chapter. For example, as corrosion products form on the surfaces of the KBS-3H steel components external to the canisters, the volume occupied by the buffer will be reduced because of the higher volume occupied by the corrosion products compared with the original metal, and buffer swelling pressure will rise. However, scoping calculations and qualitative arguments given in Appendix B.4 of the Evolution Report indicate that the rise in swelling pressure will not be so high that the safety functions of the buffer are compromised, i.e. buffer safety function indicator criteria will still be met.

Table 7-1. Issues with aspects with different significance to, or potential impact on, KBS-3H compared with KBS-3V with the potential to perturb repository safety functions, and summary descriptions of their relevance to system evolution, major uncertainties and evaluation of their implications for canister failure modes and timing and for radionuclide transport. PR: Process Report /Gribi et al. 2007/, ER: Evolution Report /Smith et al. 2007b/.

Feature/process	Relevance to system evolution	Major uncertainties	Evaluation of impact	
			Impact on canister failure mode/ timing	Impact on radionuclide transport (see Chapter 9)
Piping and erosion during the operational phase and during saturation (cf. PR Section 4.5.2; ER Section 5.5.6)	May locally perturb buffer density and increase rate of diffusion of corrosive agents to canister surface and rate of radionuclide diffusion from failed canister	Likelihood of occurrence; amount of bentonite conveyed by piped water; degree of homogenisation after piping/erosion ceases	Scoping calculations in ER Appendix B.7	Illustration of impact of increased radionuclide diffusion rates in buffer in assessment case **PD-HIDIFF**
Processes due to the presence of steel components (external to canister) and their corrosion products (cf. PR Section 4.7.1; ER Sections 5.4.2; 5.6.4; 6.5.3)	May result in chemical alteration of buffer and consequent changes to physical properties; may perturb mass transfer at buffer/rock interface	Degree and spatial extent of perturbation	Scoping calculations in ER Appendix B.7 (impact on capacity of buffer to protect canister in the event of rock shear movements < 10 cm assumed to be negligible)	Illustration of impact of increased radionuclide mass transfer at buffer/rock interface and mixing in outer part of buffer in assessment cases **PD-FEBENT1**; **PD-FEBENT2**; **PD-FEBENT3**
	May provide sorbing surfaces for radionuclides (which could be released following a change in groundwater chemistry); Fe(II) may compete for sorption sites on buffer; it may also act as a sink for dissolved sulphide, reducing the flux of sulphide to the canister surface.	Quantitative understanding of impact; possibility of release of sorbed radionuclides in the event of change in groundwater chemistry	Favourable effect of iron acting as sink for sulphides not evaluated quantitatively	Impact on sorption not assessed (remaining issue for further study); impact on buffer as a whole of change in groundwater chemistry at 70,000 years due to influx of glacial meltwater illustrated in **PD-GWMC**

H_2 from corrosion of steel components (external to canister) (cf. ER Sections 5.3.1; 5.6.4; 5.7.4)	May delay saturation in tight drift sections	Quantitative understanding of impact	None expected	None expected
	May participate in microbial reduction of sulphate to sulphide, which may subsequently corrode canister surface	Quantitative understanding of impact	Scoping calculations in ER Appendix B.7 (minor impact)	None expected
	May perturb groundwater flow and radionuclide transport in the geosphere for the first few thousand years	Quantitative understanding of impact	Minor impact on mass transfer of corrosive agents between geosphere and buffer (not quantitatively evaluated)	Impact on radionuclide transport for an initially defective canister not assessed (remaining issue for further study)
High-pH leachates from cementitious components (cf. ER Section 5.6.5)	May result in chemical alteration of buffer and consequent changes to physical properties; may perturb mass transfer at buffer/rock interface	Degree and spatial extent of perturbation	Scoping calculations in ER Appendix B.7	Illustration of impact of increased radionuclide mass transfer at buffer/rock interface and mixing in outer part of buffer in assessment cases **PD-FEBENT1**; **PD-FEBENT2**; **PD-FEBENT3**
KBS-3H drift and surrounding EDZ/rock spalling (cf. ER Sections 4.1.2; 5.4.5)	May perturb mass transfer at buffer/rock interface	EDZ hydraulic properties; impact of buffer swelling on rock spalling; transport characteristics of spalled zone	Scoping calculations in ER Appendix B.7	Illustration of impact of rock spalling in assessment case **PD-SPALL**
Expulsion of water from a defective canister interior by gas (cf. PR Section 2.5; ER Section 8.10.3)	Expelled water may convey dissolved radionuclides	Location of canister defect; rate of gas pressure build-up inside canister	No major effects (minor impact on likelihood of canister failure by rock shear; ER Appendix B.5)	Impact of expulsion of contaminated water by gas illustrated in assessment case **PD-EXPELL**

7.1.2 Piping and erosion

a. General description

The distance blocks in the current reference design have been the subject of extensive experimental and modelling studies to evaluate whether or not significant piping and erosion might occur during the heterogeneous saturation of the drift (Appendix L of /Autio et al. 2007/). These studies have been used to guide repository design and, in particular, the dimensioning of the distance blocks and the setting of drift inflow criteria to avoid the possibility that significant piping and erosion will occur (Section 4.2). In the current reference design, the distance block has been shown experimentally not to undergo piping even under pessimistic assumptions of water inflow and hydraulic pressure build-up. On the basis of these studies, it is concluded that piping and erosion are unlikely to occur in the reference design given the current criterion on the maximum initial inflow rate to an ~ 10 m long drift section used for supercontainer and distance block emplacement of 0.1 litres per minute. This conclusion is, however, dependent on the assumption that hydraulic pressure differences between the water-filled gaps around the supercontainers in less tight drift sections and other, tighter parts of the drift cause no significant mechanical deformation or displacement of the distance blocks.

In the current reference design, displacement of the distance blocks is prevented during compartment operation by the fixing rings that are installed along the drift and, following operations, by the compartment plug, which is designed to withstand full hydrostatic pressure at repository depth. A critical issue, however, is the possibility that even a small deformation or movement of a distance block relative to an adjoining supercontainer could create an open gap along the vertical face of the block. Hydraulic pressure would then be exerted on the full surface area of the face, rather than on a narrow annulus region close to the drift wall. In this situation, the fixing rings would not be adequate to prevent a more serious movement of the distance block, which would then increase the likelihood of piping along the bentonite/host rock interface. As noted in Section 11.4.1, further design developments are being considered to improve the robustness of the engineered barrier system with respect to possible distance block deformation and displacement by hydraulic pressure differences.

Even if piping were to occur, the limited duration of flow before the downstream void spaces became water filled would limit its potential to cause redistribution of buffer mass (at least once a compartment had been plugged). Assuming rapid homogenisation of any local density reductions, the scoping calculations presented in Appendix B.4 of the Evolution Report indicate that redistribution of buffer mass is unlikely to lead to buffer densities, swelling pressures or hydraulic conductivities outside the ranges set by the safety function indicator criteria given in Table 4-2. Thus, transport of dissolved species within the buffer is expected to remain diffusion dominated, although the diffusion coefficient of the buffer could conceivably be increased to some extent, as assumed when evaluating the potential impact of piping and erosion on canister lifetime and on radionuclide release and transport.

It should also be noted that there are significant uncertainties in the degree to which the buffer would homogenise following significant localised erosion. Homogenisation will be resisted by internal friction within the buffer and friction between the buffer and fixed surfaces. /Börgesson and Hernelind 2006/ carried out a modelling study of the homogenisation of a KBS-3V buffer, including the resealing of pipes and buffer swelling following a local loss of buffer mass, e.g. due to piping. The calculations showed that, due to friction, locally decreased densities and swelling pressures and increased hydraulic conductivities will persist indefinitely. Thus, avoidance of significant piping remains a critical issue in repository design.

b. Impact on canister lifetime

The interaction of the copper canister with sulphide from the groundwater is considered to be the most important canister corrosion process in the long term, after oxygen-free conditions have become established in the drift. The corrosion rate is limited by the slow rate of diffusion of sulphide across the buffer to the canister surface. However, scoping calculations presented in Appendix B.7 of the Evolution Report, the results of which are shown in Figure 6-4, indicate that canister lifetime is only slightly reduced if an increased sulphide diffusion coefficient in the buffer is assumed, e.g. due to the loss of buffer material by piping and erosion during early evolution. Calculations of the swelling and homogenisation of the buffer following a local loss of bentonite carried out in the context of a KBS-3V repository indicate that transport in the buffer will remain diffusion dominated even following the erosion of a few hundred kilograms of material, /see Börgesson and Hernelind 2006/. In the absence of other more significant perturbations to mass transport in the buffer or at the buffer/rock interface, piping and erosion are therefore judged not to have the potential to lead to canister failure by copper corrosion in a million year time frame.

c. Impact on radionuclide release and transport

An increase in buffer diffusion coefficient due to piping and erosion could potentially perturb radionuclide transport across the buffer in the event of canister failure. A calculation in which an increased buffer diffusion coefficient is assumed is presented in Chapter 9 in order to illustrate the impact on radionuclide release and transport (assessment case PD-HIDIFF).

7.1.3 Steel components external to canister, their corrosion products and their impact on mass transport

a. General description

The presence of porous or fractured supercontainer shell corrosion products at the drift wall and also mineralogical transformation of the buffer due to iron/bentonite interaction each have the potential to perturb the mass transfer properties of the buffer/rock interface, and thus affect both canister lifetime and radionuclide transport in the event of canister failure.

The porosity and hydraulic conductivity of magnetite, the expected initial corrosion product, formed under repository conditions is uncertain, and may be very low. It may, however, be fractured and the possibility of it forming a hydraulically conductive layer at the buffer/rock interface cannot currently be excluded. The impact of mineral transformation on the hydraulic and rheological properties of the transformed clay is also uncertain. In the zones where transformation occurs, there is the possibility of a substantial loss of swelling capacity, an increase in brittleness and a significant increase in hydraulic conductivity. These zones are, however, likely to remain hydraulically isolated from each other by the distance blocks, which will be largely unaffected by mineral transformation because they are spatially well separated from the steel components.

In calculations in the present safety assessment that consider the potential impact of these phenomena on canister lifetime and on radionuclide release and transport subsequent to canister failure, it is assumed that they give rise to a spatially limited but highly hydraulically conductive zone at the buffer/rock interface. In reality, the hydraulic conductivity of the affected zone may remain very low, and this is the assumption, for example, in the majority of calculations of radionuclide release and transport not directly addressing the issue of iron/bentonite interaction and other perturbations to the buffer/rock interface. The impact of iron/bentonite interaction in particular on the transport properties of the buffer is an area for further investigations. This interaction may affect the possibility of canister failure by corrosion in the very long term (after 100,000 years, see below), and the transfer of radionuclides across the buffer/rock interface in the event of canister failure.

b. Impact on canister lifetime

The impact of perturbations to the buffer/rock interface on the canister corrosion rate is assessed in the scoping calculations reported in Appendix B.7 of the Evolution Report /Smith et al. 2007b/, the results of which are shown in Figure 6-4. On the basis of these results, the presence of a perturbed buffer/rock interface is judged not to have the potential to lead to canister failure by copper corrosion before about 100,000 years, but may lead to canister failure by corrosion in a million year time frame in the case of canisters located near to more transmissive fractures, particularly if the sulphide concentration at the buffer/rock interface is significantly increased, e.g. by microbial activity leading to the reduction of sulphate in the groundwater to sulphide.

A potentially favourable effect not included in the scoping calculations (also noted in Section 6.2.6) is that some of the sulphide entering the buffer will react with the supercontainer and other steel components, and with amorphous iron present in the bentonite. The resulting iron sulphides have a low solubility and will precipitate, reducing the flux of sulphide to the canister surface.

Failure of a single canister by corrosion at 100,000 years is considered in radionuclide release and transport calculations addressing this particular canister failure mode (assessment cases with names beginning with CC-, see Chapter 9).

c. Impact on radionuclide release and transport

The presence of a hydraulically conductive zone at the buffer/rock interface could perturb radionuclide release from the buffer to the geosphere in the event of canister failure, irrespective of the failure mode. Calculations of radionuclide release and transport for a canister with an initial penetrating defect in which the buffer/rock interface is treated as a highly conductive mixing tank are presented in Chapter 9 (assessment cases PD-FEBENT1, PD-FEBENT2 and PD-FEBENT3).

Another possible effect is that Fe(II) sorption on buffer pore surfaces may locally compete with that of some radionuclides released subsequent to canister failure, and thus weaken the barrier function of the buffer for these radionuclides, which include, for example, Ni. Competitive sorption of Fe(II) is currently not well understood. There is a lack of experimental data and the phenomenon is not treated quantitatively in the present safety assessment. Although reduction in sorption has only a minor effect on the release of radionuclides that are solubility limited (except where the half-lives are relatively short), competitive sorption of Fe(II) is identified as an issue that would benefit from further work.

7.1.4 Impact of gas from the corrosion of steel components external to canister

a. General description

Hydrogen gas generated by the corrosion of the supercontainer steel shell and other steel components external to the canisters may affect both the saturation of the repository and, via its effect on the microbial reduction of sulphate to sulphide and the possible effect on the bentonite porewater chemistry, the corrosion of the canisters. Furthermore, as gas rises through the fracture network within the host rock, it may carry water with it, and perturb groundwater flow to a degree not currently fully understood. The largest contribution to hydrogen gas generation comes from the supercontainer shell (there are smaller contributions from other steel components external to the canisters, i.e. the fixing rings and spray and drip shields). Assuming the shell to be corroded from both sides, the hydrogen generation period is 4,000 years for an assumed corrosion rate of 1 µm per year, and 40,000 years for a corrosion rate of 0.1 µm per year (uncertainties in the steel corrosion rate are discussed in Section 5.7.1 of the Process Report).

The degree to which gas can accumulate in a given drift section, and hence the degree to which it affects groundwater inflow and drift saturation, depends strongly on the hydraulic properties of the adjacent rock, including the excavation damaged zone (EDZ) around the drift and transmissive fractures in the rock that intersect the drift wall (Section 6.2.4). However, in tight drift sections in which the average hydraulic conductivity of the rock is in less than about 10^{-13} m s^{-1}, repository-generated gas from the corrosion of steel components in the drift is expected to hinder or prevent altogether the saturation of the buffer until gas generation by steel corrosion ceases and gas pressure falls, which, as noted above, is expected to take up to tens of thousands of years, assuming a rate of steel corrosion rate towards the low end of the range of uncertainty.

b. Impact on canister lifetime

It is shown in scoping calculations in Appendix B.7 of the Evolution Report that, as long as hydrogen production persists, there is more than enough hydrogen present to facilitate microbial reduction of sulphate in the groundwater to sulphide, even for a low corrosion rate of 0.1 μm per year. Some of this sulphide will migrate to the canister surface, giving rise to a period of enhanced corrosion. However, the scoping calculations also show that, even in the case of a pessimistically modelled perturbed buffer/rock interface, an overall canister lifetime of several hundred thousand years is expected. Only in the case of a high sulphide concentration being maintained at the buffer/rock interface, e.g. by microbial activity, over a period in excess of the period of hydrogen gas generation by steel corrosion is canister lifetime reduced below about 100,000 years. As noted in Section 5.1.4, methane and hydrogen naturally present in the groundwater could, in principle, play a role in sustaining microbial activity over a longer period. This is an issue for both the KBS-3H and KBS-3V repository alternatives that requires further investigation (Chapter 11). The timing and likelihood of canister failure due to corrosion is considered further in Section 8.5.

The impact of hydrogen gas on the bentonite porewater chemistry, e.g. on the acid-base equilibria, has not been fully evaluated is noted as an issue for further study in Section 11.2.6.

c. Impact on radionuclide release and transport

In most drift sections, the buffer is expected to be largely saturated by the time of peak radionuclide releases from a failed canister. As discussed in Chapter 9, in the case of an canister with an initial penetrating defect, early releases are relatively small, and the maximum only occurs once the transport resistance of the defect is lost, which may take up to several thousands of years. However, in the drift sections with an average wall-rock hydraulic conductivity less than about 10^{-13} m s^{-1} , depending on the rate of steel corrosion, the buffer may remain only partly saturated at this time (see *a*, above). The impact on radionuclide release to the geosphere has not as yet been quantified, but releases are expected to be no more than in the case of a fully saturated buffer, and may be somewhat reduced. Any perturbation to groundwater flow in the geosphere due to the gas rising through the fracture network is also likely to have largely ceased by the time most radionuclides are released from failed canisters, except possibly in the tightest drift sections where groundwater flow is in any case virtually zero. Thus, no calculations of radionuclide release and transport dealing with the impact of gas from the corrosion of steel components external to the canister are considered in the present safety assessment.

7.1.5 Interactions involving leachates from cementitious components

a. General description

Cementitious repository components will, over time, produce alkaline leachates that will react with the host rock and with the buffer. In the deposition drifts, it is currently foreseen that low-pH cement (or colloidal silica) will be used. This is expected to have reduced detrimental effects on the host rock and buffer compared with Ordinary Portland Cement because of the reduced alkalinity of their porewater leachates. Nevertheless, reacting minerals will still dissolve and new minerals will be precipitated, creating a zone of altered mineralogy around the cementitious components. Quantitative understanding of these processes is limited, with poor knowledge of the possible reaction products and of mineral dissolution and growth kinetics. When the leachates reach the buffer, the most significant potential impact will be the dissolution of montmorillonite. Secondary minerals could also dissolve or precipitate in the buffer. The reaction products are likely to be amorphous with no expandability, but their porosity is low with very low hydraulic conductivity. The long-term safety concern is the possibility of a loss of buffer swelling pressure, increased hydraulic conductivity, and possibly fracturing of bentonite at cement/bentonite interfaces due to cementation.

Mass balance calculation results (Appendix F in the Process Report, /Gribi et al. 2007/) indicate that the maximum mass of bentonite altered by cement in a KBS-3H drift is small – in the order of 1‰, assuming that all the cementitious material used for grouting of the rock in ONKALO and in the repository below a depth of 150 m is distributed evenly amongst the repository drifts. There could, however, be more significant local effects on the buffer. Potentially, the most significant scenario of cement/bentonite interaction occurs through indirect contact of alkaline cementitious porewater transported from a grouted fracture by groundwater to the supercontainer area through the fracture network. Conservative mass balance and mass transport calculation results indicate that the thickness of the altered buffer zone can be up to 4 cm at most.

b. Impact on canister lifetime

The above discussion indicates that exposure of the canisters to high-pH leachates is not expected, although there remain significant uncertainties. However, even if exposure were to occur, it is likely to have a favourable impact on corrosion of the canister surfaces, through the production of a passivating film /King 2002/.

It is more likely that canister lifetime is affected indirectly, by changes to the mass transport properties of the outer part of the buffer and the buffer/rock interface. Calculations in the present safety assessment that consider the potential impact of leachates from cementitious components on canister lifetime are identical to those addressing the effects of steel corrosion products and iron/bentonite interaction, which also may perturb the mass transfer properties of the buffer/rock interface.

As noted in Section 7.1.3, the presence of a perturbed buffer/rock interface is judged not to have the potential to lead to canister failure by copper corrosion before about 100,000 years, but may lead to canister failure by corrosion in a million year time frame in the case of canisters located near to more transmissive fractures, particularly if the sulphide concentration at the buffer/rock interface is significantly increased, e.g. by microbial activity.

c. Impact on radionuclide release and transport

Calculations in the present safety assessment that consider the potential impact of leachates from cementitious components on radionuclide release and transport are also identical to those addressing the effects of steel corrosion products and iron/bentonite interaction. No specific calculations addressing the presence of highly alkaline cementitious waters within and around the repository have been performed, although the assessment cases that consider glacial meltwater cover a situation with fairly high pH conditions. Calculations of radionuclide release and transport for a canister with an initial penetrating defect in which the buffer/rock interface is treated as a highly conductive mixing tank are presented in Chapter 9 (assessment cases PD-FEBENT1, PD-FEBENT2 and PD-FEBENT3).

7.1.6 Thermally induced rock spalling

a. General description

Spalling is the brittle fracturing of rock surface into splinters, chips or fragments. Thermo-mechanical analyses of a KBS-3H repository at Olkiluoto by /Lönnqvist and Hökmark 2007/ indicate that, in the reference layout in which the deposition drifts are aligned with the direction of the principal stress in the rock, little or no spalling will occur prior to the emplacement of spent fuel. Significant thermally-induced spalling could, however, occur on a timescale of a few years, although it is most likely around supercontainers in relatively tight drift sections, since in less tight drift sections and around the relatively tightly fitted distance blocks, buffer swelling pressure is likely to stabilise the drift wall before spalling occurs (Section 5.4.5 if the Evolution Report, /Smith et al. 2007a/). The use of pellets to prevent thermally-induced spalling is being studied in the context of the KBS-3V design, and such measures may also be considered in future KBS-3H design studies (Chapter 11).

b. Impact on canister lifetime

Calculations in the present safety assessment that consider the potential impact of thermally-induced rock spalling are identical to those addressing the effects of steel corrosion products, iron/bentonite interaction and leachates from cementitious components, which may also perturb the mass transfer properties of the buffer/rock interface.

c. Impact on radionuclide release and transport

A calculation of radionuclide release and transport for a canister with an initial penetrating defect in which the buffer/rock interface is perturbed by thermally-induced rock spalling and is treated as a highly conductive mixing tank is presented in Chapter 9 (assessment case PD-SPALL). The assumed groundwater flow in the perturbed interface region takes into account the fact that thermally-induced spalling is considered most likely in relatively tight drift sections, and is lower than that assumed in cases covering the effects of steel corrosion products, iron/bentonite interaction and leachates from cementitious components on the interface (assessment cases PD-FEBENT1, PD-FEBENT2 and PD-FEBENT3). However, given that there is uncertainty in the buffer swelling pressure required to stabilise a drift section against thermally-induced spalling, the possibility of spalling in less tight sections cannot currently be excluded, and so these cases are also considered to be relevant to the spalling issue.

7.1.7 Expulsion of water and dissolved radionuclides from a defective canister interior by gas

a. General description

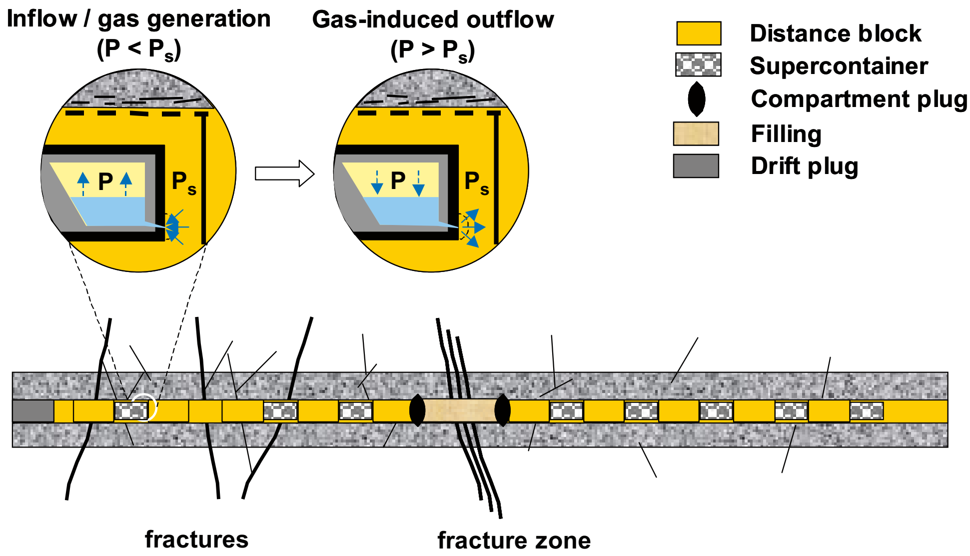
Water ingress to a defective canister and subsequent expulsion of water and dissolved radionuclides by gas generated and trapped inside the canister is a possibility for both the KBS-3H and KBS-3V designs. It is, however, more likely in the case of a KBS-3H repository compared with KBS-3V if it can be assumed that penetrating defects are a possibility primarily in the welding region, which is located at the top of the canister. If this is the case then, in KBS-3V, gas can escape from a vertically orientated defective canister without the development of pressures that could expel water and radionuclides. In KBS-3H, on the other hand, the defect could be located on the lower side of the horizontally orientated canister, which would allow gas to become trapped (Figure 7-1). The gas pressure inside the canister will then rise until it exceeds the gas breakthrough pressure of the bentonite. Gas pathways will form that may transport radionuclides present as volatile species (mainly C-14). The gas pathways will remain open until gas production ceases or is greatly reduced.

b. Impact on canister lifetime

The issue relates to a situation where the canister has already failed, and so there is no impact on canister lifetime.

c. Impact on radionuclide release and transport

Model calculations elucidating the conditions under which gas-induced displacement of radionuclide-contaminated water from the canister interior through a penetrating defect in the canister shell may occur are described in the Process Report. Results indicate that the more likely situation is that water entering the canister will be completely consumed by corrosion of the cast iron insert, and there will be no gas-induced displacement of contaminated water



Note: the amount of free gas (light yellow) within the canister is changed by a number of different processes (gas generation, advection and diffusion of dissolved gases, dissolution/degassing).

Figure 7-1. *Conceptual model for transport of water and gas into and out of a canister with an initial penetrating defect /after Gribi et al. 2007/.*

through the defect into the saturated bentonite. The possibility of expulsion of contaminated water by gas cannot, however, be completely excluded, and its impact on radionuclide release and transport is addressed in a radionuclide release and transport calculation presented in Chapter 9 (assessment case PD-EXPELL).

7.2 Key issues in the longer-term evolution of a KBS-3H repository

7.2.1 Identification of issues in SR-Can

In its safety assessment of a KBS-3V repository at two potential Swedish sites, the SKB safety assessment SR-Can concluded that no canister failures, and therefore no releases of radionuclides, are expected for either of the Swedish sites considered during the initial temperate period after deposition, expected to last for several thousand years. Thereafter, it was further noted that the most severe safety issues concern future glacial periods, and a number of conclusions regarding these issues were drawn. Similar conclusions were drawn in Posiva's KBS-3V Evolution Report /Pastina and Hellä 2006/. These conclusions are set out in the following sections, and comments made regarding their relevance to a KBS-3H repository at Olkiluoto.

7.2.2 Buffer freezing

Freezing of the buffer as a result, for example, of the deep penetration of permafrost following a major climate change would, if it were to occur, lead to detrimental changes in buffer properties that could compromise its capacity to protect the canister and to limit and retard radionuclide releases from a failed canister. As noted in Section 12.4.1 of /SKB 2006a/, it is uncertain what transport properties the buffer would have after thawing.

In SR-Can, freezing of an intact buffer was assessed as ruled out for both of the sites considered, even for the most pessimistic assumptions regarding future climatic conditions. In the case of a KBS-3H repository at Olkiluoto, according to present knowledge based on past glaciations, any future permafrost layer is not expected to reach more than 180 metres below ground at Olkiluoto

/Hartikainen 2006/ and is thus not considered as a potential cause of major loss of buffer safety functions in the present safety assessment. The possibility that conditions at Olkiluoto could in the future differ significantly compared with those during the past glaciations and lead to buffer freezing may, however, require further consideration in future studies.

7.2.3 Canister failure due to isostatic load

Glacial loading will greatly increase pore pressures in the rock and in the saturated buffer, and hence the isostatic load exerted on the canisters. In SR-Can, however, canister failure due to isostatic load was assessed as ruled out for both of the sites considered, again even for the most pessimistic assumptions regarding future climatic conditions. In the case of a KBS-3H repository at Olkiluoto, collapse due to isostatic loading is also ruled out (see Sections 5.4.3 and 7.4.4 of the Evolution Report). At Olkiluoto, the maximum expected isostatic load on the canister in the absence of an ice sheet is about 11–12 MPa, which is the sum of the swelling pressure of the bentonite and the hydrostatic pressure at repository depth. The maximum ice thickness over Olkiluoto is estimated to have been about 2 km during the last glacial maximum /Lambeck et al. 1998/. A 2 km-thick ice sheet will increase the load on the canisters by about 18 MPa and a 3 km thick ice sheet by about 27 MPa (/SKB 1999c, Rasilainen 2004/; assuming an average ice density of 900 kg m^{-3}). The expected minimum pressure giving rise to total collapse of an intact canister is significantly higher than this (80–114 MPa – see, e.g. Section 6.2.3 of /Pastina and Hellä 2006/), and so no failure of canisters is expected by this mechanism without prior weakening of the insert by corrosion (which could only occur if the canister has already failed due, for example, to the presence of an initial penetrating defect). A more localised failure may be possible at lower pressures, but is considered to have a negligible probability (Section 7.4.4 of the Evolution Report, /Smith et al. 2007b/). Additional rock stresses due to glacial loading could potentially result in creep movement and possible convergence of repository drifts, which would in turn affect the density and the swelling pressure of the buffer, and indirectly give rise to some further loading on the canisters. In /Rasilainen 2004/, however, it is stated that time-dependent deformations (creep) are not considered to be a significant factor for intact crystalline rock in the case of a KBS-3V repository, and the possibility of creep deformation is not considered to be a significant issue in either SR-Can or the present safety assessment.

7.2.4 Oxygen penetration to repository depth

The migration of oxygen to repository depth in association with future glacial cycles, if it were to occur, could increase canister corrosion rates. In SR-Can, oxygen penetration to repository depth was assessed as ruled out for both the sites considered, in agreement with the conclusions of earlier assessments by SKB. This conclusion is, however, regarded as preliminary, given that it is based on simplified and stylised modelling. In the case of Olkiluoto, although glacial meltwater may penetrate to repository depth, the recent interpretation of hydrogeochemical site data, and especially isotopic gas data, by /Pitkänen and Partamies 2007 and Andersson et al. 2007/ gives no evidence for intrusion of oxygen to repository depth in the past (Section 7.3.5 of the Evolution Report, /Smith et al. 2007b/). This can be attributed to the consumption of oxygen by microbially-mediated reactions and the interaction of oxygen with minerals in the rock. Oxygen intrusion to the repository depth would imply extensive flushing of the groundwater system at this depth with glacial meltwater. Current understanding of the history of the Olkiluoto site, however, rather suggests only moderate dilution of groundwater as a result of meltwater intrusion. Based on these arguments, the future intrusion of oxygen-rich glacial meltwater into the deep groundwater system at Olkiluoto is unlikely. Furthermore, even if this were to occur, scoping calculations by /Ahonen and Vieno 1994/ indicate that canister failure by corrosion would hypothetically require exposure to oxygenated water to be maintained for at least 100,000 years. Thus, in the present safety assessment, the possibility of failure by corrosion due to oxygen penetration to repository depth is excluded. It should also be noted that, even if oxygen-rich glacial meltwater were to reach the repository, an additional scavenger for oxygen in a KBS-3H repository of the current reference design would be the iron in divalent form originating from the corrosion of the steel supercontainer shell.

7.2.5 Canister failure due to rock shear in the event of a large earthquake

In both the KBS-3V and KBS-3H designs, shear movements on fractures in the event of a large earthquake could lead to a deformation of the bentonite buffer and to additional stresses being exerted on the canisters. Seismic activity at the Swedish sites considered in SR-Can and at Olkiluoto is currently low. However, canister failures due to rock shear associated with post-glacial earthquakes were not completely ruled out in SR-Can safety assessment and cannot be ruled out in the present safety assessment for a KBS-3H repository at Olkiluoto.

The buffer is expected to protect the canister against rock shear movements of the order of 0.1 m and smaller with a significant safety margin /Börgesson et al. 2004/. The maximum amount of movement is related to fracture size. There is, however, uncertainty in the degree to which large fractures with the potential to slip by more than 0.1 m can be identified and avoided when emplacing canisters along the drift in both designs, and so the potential consequences of shear movements greater than 0.1 m at canister emplacement locations need to be assessed.

In SR-Can, the risk contribution from this failure mode was assessed to be small, with probabilistic analyses implying that, on average, it would take considerably more than a million years for even one such failure to occur. The calculated probability of canister failure is significantly reduced in SR-Can by assuming that the Expanded Full Perimeter Criterion (EFPC) is applied, whereby large fractures intersecting both the full perimeter of a KBS-3V deposition tunnel and the deposition hole are assumed to be readily observable and avoided /Munier 2006/.

The timing and likelihood of canister failure due to rock shear for a KBS-3H repository at Olkiluoto is considered in Section 8.6. No criterion similar to the EFPC is applied, since relatively small fractures could intersect the full perimeter of a KBS-3H drift, and, if all such fractures are avoided, this could, in principle, render large parts of the drifts unsuitable for canister emplacement. This has not, however, been tested in practice and the applicability of a modified form of the EFPC criterion remains, however, an issue for further study (Section 11.4.4).

7.2.6 Loss of buffer from exposure to glacial meltwaters and implications for canister lifetime

Penetration of dilute glacial meltwater to repository depth, if it were to occur, would lead to an increase in buffer swelling pressure and, if the concentration of cations in solution falls below the Critical Coagulation Concentration (CCC), to the break-up and dispersion in the form of colloids of buffer material that has been extruded into fractures intersecting the repository drift. This phenomenon, which is termed "chemical erosion", is discussed in detail in Section 2.5.10 of /SKB 2006c/. The cation concentration at the interface is determined by the cation concentration in the groundwater, and also by the supply of cations to the interface through the dissolution of Ca and Mg minerals in the buffer and subsequent diffusion of these ions to the interface. The CCC is uncertain but, according to Section 2.5.10 of /SKB 2006c/, is around 1 mmol per litre in the case of divalent ions. Calculations of chemical erosion in a KBS-3V repository presented in SR-Can suggest that the possibility of advective conditions becoming established at some locations in the buffer cannot be excluded, and that this may occur as early as the next period of glaciation in the least favourably located deposition holes /Section 12.7 of SKB 2006a/. The establishment of advective conditions in the buffer would increase the rate at which sulphide from the groundwater could migrate to the copper surface, and hence increase the copper corrosion rate and reduce canister lifetime.

In SR-Can, it is concluded that, in a one million year perspective, chemical erosion could lead to failure of up to some tens of canisters at the Forsmark site, although the calculations of chemical erosion are tentative and subject to significant model uncertainties. It is acknowledged in SR-Can that understanding of buffer erosion under relevant conditions is an area warranting further research, and that better understanding of the erosion process could lead to models that yield either lower or higher erosion rates and higher or lower rates of canister failure.

In the case of Olkiluoto, while the future intrusion of oxygen-rich glacial meltwater into the deep groundwater system is considered unlikely, a transient reduction in salinity in association with glacial retreat is considered possible. Ongoing site characterisation work is exploring whether or not such events have happened in the past. The timing and likelihood of canister failure due to corrosion following penetration of glacial meltwater to repository depth and subsequent chemical erosion of the buffer or significant perturbation to the buffer/rock interface is considered in Section 8.5.

A related issue specific to KBS-3H is that, sufficiently extensive, mineral transformation of the buffer by iron/bentonite interaction and the associated loss of buffer plasticity could make the canister more vulnerable to failure by rock shear in the event of a large earthquake. According to /Börgesson et al. 2004/, the unperturbed buffer is expected to protect the canister against rock shear movements of the order of 10 cm and smaller with a significant safety margin. Since, in reality, somewhat larger shear movements are also unlikely to result in canister damage in the case of an unperturbed buffer, and since mineral transformation is only expected to affect a small part of the buffer near to its interface with the rock /Wersin et al. 2007/, the capacity of the buffer to protect the canisters from rock shear movements smaller than 10 cm is expected to be maintained in the event of such transformation.

7.2.7 Implications of a prolonged period of temperate climate (greenhouse effect)

Prior to any future glaciation, continuing isostatic uplift will result in the gradual replacement of the saline water at repository depth with more brackish water of lower ionic strength. As noted in Section 5.1.6, the land uplift rate is expected to vary little over the next few centuries, but will decrease significantly within the next few thousand years. Nevertheless, prolonged temperate conditions without glaciation, which could arise as a result of anthropogenic emissions, and especially greenhouse gases, will result in more prolonged uplift and a longer period of seepage of dilute surface water towards repository level compared with the case in which no significant climatic warming due to such anthropogenic effects takes place. In the main report of SR-Can, SKB concludes that the groundwater conditions during the post-closure temperate phase under the greenhouse variant of SKB's reference evolution scenario will be similar to those of the reference evolution, with the difference that a longer period of exposure to dilute groundwaters of meteoric origin is expected at repository depth /SKB 2006a/. Nevertheless, compared with glacial meltwater penetration, surface water infiltration is a slow process because it is driven only by the hydraulic gradient created by the residual land uplift, which, in the case of the Olkiluoto site, is expected to be some tens of metres in the next few tens of thousands of years. Slowly infiltrating surface water will interact with rock minerals and mix with deeper, more saline groundwater layers. This, together with organic activity in the soil layers, will increase its ionic strength. The Critical Coagulation Concentration (CCC) is, therefore, expected to continue to be exceeded irrespective of any climatic warming due to fossil fuel burning. Although uncertainty remains and the issue is noted in Chapter 11 as requiring further investigation, the stability of the bentonite gel/water interface is expected to be maintained, and no significant erosion of the buffer is expected to occur prior to any future change to glacial conditions and the penetration of glacial meltwater to repository depth. More generally, it is concluded in SR-Can that, since the processes that are potentially the most detrimental to repository safety are related to glacial conditions (Section 7.2.2 to 7.2.6), a prolonged period of continuing temperate conditions (Section 7.2.7) would be generally beneficial for safety. The same conclusion applies to a KBS-3H repository at Olkiluoto, although further evaluations of geochemical evolution in this scenario would be beneficial to reinforce this conclusion.

8 Base scenario and scenarios leading to canister failure and radionuclide release

8.1 Methodology to identify scenarios

In view of the safety issues described in Chapter 7, there are scenarios leading to canister failure and radionuclide release within a one million year time frame that need to be identified and analysed. In the present safety assessment, the methodology used to identify scenarios is taken directly from SR-Can. It can be described in terms of the following steps:

1. Consider the safety functions of each of the main components of the disposal system.
2. For each safety function, identify one or more safety function indicators.
3. For each safety function indicator, derive safety function indicator criteria.
4. Develop understanding of the system and its evolution – with a focus on the safety functions.
5. Identify the failure modes (loss of safety functions) that could occur in the course of system evolution.
6. Consider if and when the occurrence of such failure modes is plausible.
7. Consider the implications of loss of one safety function on the others.
8. Identify plausible descriptions of the evolution of safety functions over time.

Steps 1-7 are covered by the earlier chapters of this report. The present chapter deals with Step 8.

The products of this step – plausible descriptions of the evolution of safety functions over time – are described in the following as "scenarios", although, in SR-Can, because of the way in which Swedish regulatory guidance is formulated, all those that are considered "probable" fall within the scope of one or or both of the two variants of a single scenario (termed the "main" scenario in SR-Can) describing the reference evolution of the disposal system. Other scenarios considered in SR-Can address situations that are considered less likely to occur (see Section 11.3 of the main report of SR-Can, /SKB 2006a/).

8.2 Scenario overview

The various scenarios shown in Figure 8-1 have been identified on the basis of steps 1-7 of the methodology outlined above. The scenarios are shown as combinations of barrier states (a "no fail" state indicates that all safety functions are assumed to be fulfilled), and are grouped according to the initial canister failure mode that they involve.

As indicated by arrows in Figure 8-1, scenarios involving canister failure and radionuclide release are initiated by:

- the presence of an initial, penetrating defect in one or more of the canisters,
- perturbations to the buffer and buffer/rock interface[17], giving rise to an increased rate of transport of sulphide from the geosphere to the canister surface and an increased canister corrosion rate,

[17] The buffer state "low density/alteration (outer buffer)" includes perturbations to the buffer/rock interface that primarily affect the rock, i.e. rock spalling and the presence of a conductive excavation damaged zone (EDZ).

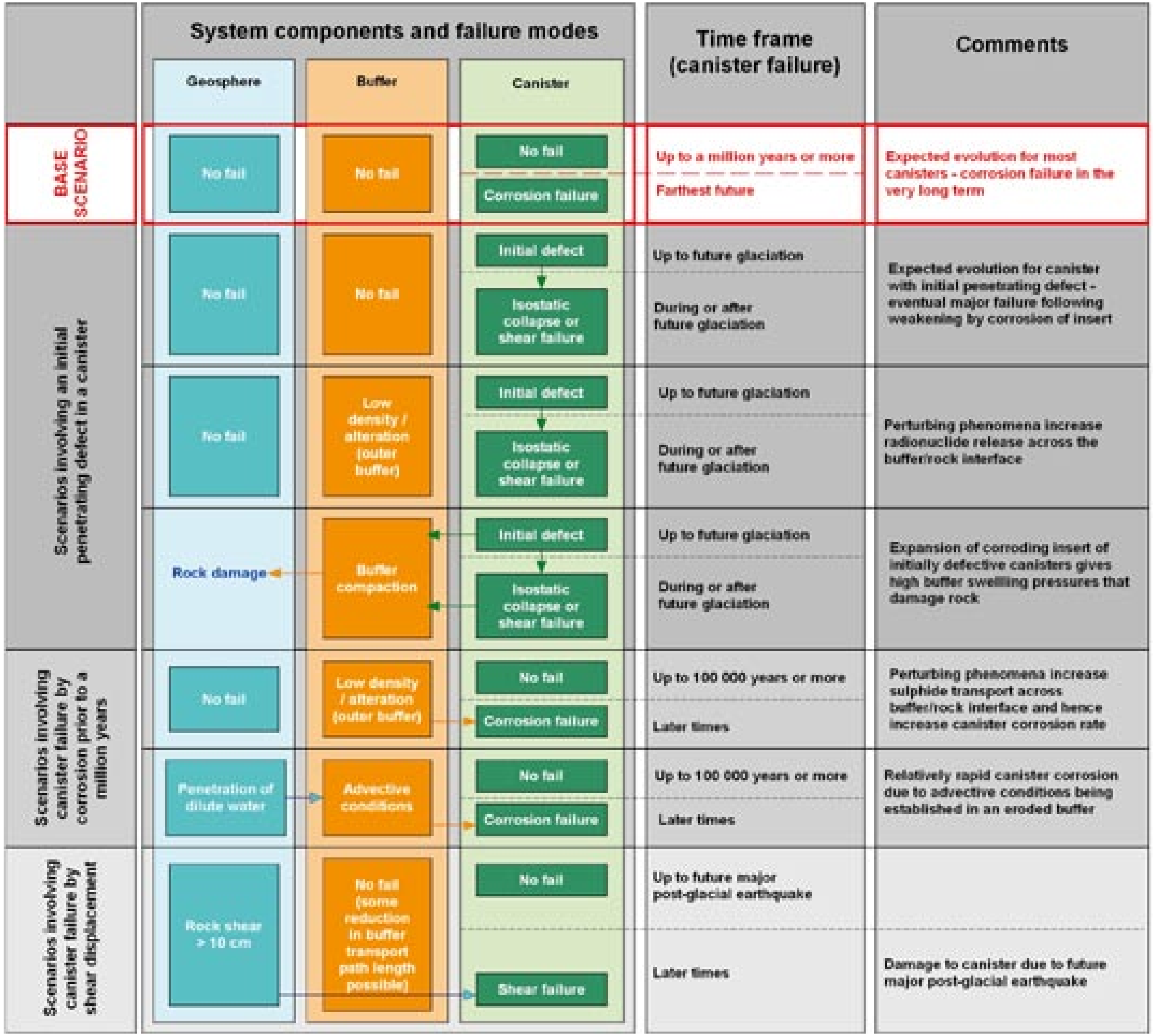


***Figure 8-1.** Potential system states in different time frames analysed in the present safety case – the Base Scenario is shown in red.*

- penetration of dilute glacial meltwater to repository depth, giving rise to chemical erosion of the buffer and an increased rate of transport of sulphide from the geosphere to the canister surface and an increased canister corrosion rate,
- rock shear movements of sufficient magnitude to give rise to shear failure of the canisters.

The Base Scenario in which there is no radionuclide release in a million year time frame is discussed in Section 8.3. Other scenarios, which involve canister failure and radionuclide release within this million-year time frame, are described in Sections 8.4 to 8.6.

As noted in Section 2.3, the importance to long-term safety of human intrusion scenarios (e.g. boring a deep water well at the disposal site and core drilling hitting a spent fuel canister) has not be evaluated, although current Finnish regulations indicate that this will be a requirement of any future safety case. Human intrusion has, however, been addressed in the main report of the Swedish SR-Can safety assessment by considering a borehole hitting a canister and then subsequently being used for drinking water abstraction /SKB 2006a/. Drilling occurs 300 years after the sealing of the repository and the calculated doses are in the range of 10^{-4} to 10^{-3} Sv per year, assuming a fuel degradation rate of 10^{-7} per year and water flow through the affected canister of 1,000 litres per year.

8.3 The Base Scenario of no canister failures within a million year time frame

In the present safety assessment, the Base Scenario for the evolution of a KBS-3H repository at Olkiluoto assumes that the main system components (the canister, the buffer and the host rock) perform the safety functions described in Section 4.3 for a period extending to a million years or more[18]. These safety functions include complete containment of radionuclides by the canister. Thus, in the Base Scenario, there are no canister failures within a million year time frame. This does not imply that the system does not evolve over time in this scenario. Corrosion of the copper canisters initially due to oxygen entrapped at the time of deposition, and, in the longer-term, due principally to sulphide present in the buffer and in the groundwater will inevitably occur and eventually lead to failure. However, provided the other components of the system fulfill their safety functions, rates will be low – in the order of a few tens of nanometres per year /Pastina and Hellä 2006/ – and canister integrity should be preserved for at least a million years.

8.4 Scenarios involving the presence of initial penetrating defects

Given the central role of the canisters in the KBS-3H and KBS-3V safety concept, the possibility of initial defects that penetrate the copper shell is of crucial importance. In principle, such defects could occur anywhere in the copper shell, but they are most likely to occur along welds and, in particular, at the seal of the canister top lid.

SKB has made a first evaluation of the reliability of the friction stir welding process for canister sealing proposed in SR-Can and concluded that "the welding process produces reproducible results which satisfy stipulated requirements on minimum copper thickness with very good margins" /Ronneteg et al. 2006/. Over the entire SKB's canister inventory (4,500 canisters), it has been estimated that 99% of all canisters have no defect larger than 10 mm and 1% of all canisters have no defect larger than 15 mm with no initial penetrating defects in the copper

[18] According to the requirements given in regulatory Guide YVL 8.4 /STUK 2001/ "the base scenario shall assume the performance targets defined for each barrier, taking account of the incidental deviations from the target values".

shells of the canisters in the repository /SKB 2006b/. SKB also studied the reliability of non-destructive testing of the copper seal weld and concluded that the proposed NDT methods are suited to indicate the weld quality /Müller et al. 2006/.

Posiva plans to seal the lid of the copper overpack with electron beam welding, with friction stir welding as an alternative option. Although there are differences in Posiva's and SKB's canister designs, including the chosen reference welding technique, the probability of an initial penetrating defect is also expected to be low in case of Finnish canisters.

In both welding methods, the seal location is on or within a few cm from the end-face of the canister so that no differences can be identified between electron beam welding and friction stir welding from the point of view of radionuclide releases if it is assumed that one or more initial penetrating defects escapses detection. Posiva does not as yet take any position on the likelihood of occurrence of defective canisters, except that it will be designed to be low. This is because, in Finland, the quality assurance program for non-destructive testing techniques is still in an early phase of development. A non-destructive examination method for canister component manufacture and sealing will be selected by the end of 2008. A qualification programme for the applicable examination procedures will be developed by the end of 2009 and executed by the end of 2012, before the application for the encapsulation plant construction license is submitted.

The evolution of a canister with a postulated initial penetrating defect is described in Section 6.5. The defective canister will initially provide some transport resistance, limiting the rate of water ingress and radionuclide release. Corrosion and volume expansion of the insert is, however, expected lead to a reduction of this transport resistance over time. It will also lead to a gradual weakening of the canister and eventual isostatic collapse or shear failure, possibly in association with a future glaciation.

Figure 8-1 depicts three scenarios involving the presence of one or more initial, penetrating defects. In all three, the canister is assumed to undergo eventual isostatic collapse or shear failure during or after a future glaciation.

- In the first scenario, the buffer and geosphere are assumed to fulfil all their safety functions for at least a million years.
- In the second scenario, the buffer/rock interface is perturbed, leading to enhanced release of radionuclides from the failed canisters across the interface. Various features and processes that may lead to perturbations of the buffer/rock interface, and that have a different significance to, or potential impact on, KBS-3H compared with KBS-3V, are described in Section 7.1. It should be noted that, while perturbations to the buffer/rock interface may also affect the rate of corrosion of the copper canister, the corrosion rate of the copper canister will still remain low (isostatic collapse or shear failure will eventually occur, as in the other scenarios involving an initial penetrating defect).
- In the third scenario, corrosion and volume expansion of the cast iron insert is assumed to lead to compaction of the buffer around the canister, and an increase in swelling pressure that damages the rock (see Section 6.5).

In the radionuclide release and transport calculations described in Chapter 9, the first scenario is considered in the Base Case[19] for an initial penetrating defect (case PD-BC). The second scenario is considered in variant assessment cases PD-SPALL, PD-FEBENT1, PD-FEBENT2 and PD-FEBENT3. The third scenario is addressed implicitly in the safety assessment by defining assessment cases in which the transport resistance of the geosphere is reduced (e.g. case PD-LOGEOR).

[19] As described in Chapter 9, a Base Case is defined for each of the three canister failure modes. On the other hand, the Base Scenario describes a situation where no canister failures occur within at least a million year time frame.

8.5 Scenarios involving canister failure by corrosion

There are two scenarios depicted in Figure 8-1 in which canister failure by corrosion occurs before a million years. These are scenarios in which:

1. the buffer/rock interface is perturbed, leading to enhanced mass transfer at the interface,
2. dilute glacial meltwater penetrates to repository depth, leading to chemical erosion of the buffer and to advective conditions becoming established in the buffer.

These scenarios lead to increased rates of corrosion due to their effects on the transport of sulphide from the groundwater to the buffer and across the buffer, and hence on sulphide concentrations at the canister surface.

Features and processes that may lead to perturbations of the buffer/rock interface, and that have a different significance to, or potential impact on, KBS-3H compared with KBS-3V, are described in Section 7.1. They are:

- thermally-induced rock spalling,
- the presence of potentially porous or fractured corrosion products in contact with the drift wall relatively tight drift sections,
- chemical interaction of the buffer with these corrosion products,
- chemical interaction of the buffer with high-pH leachates from cementitious components.

There are also other processes that could potentially alter buffer transport properties and mass transfer across the buffer/rock interface (e.g. cementation due to silica precipitation, strain due to deformation of the supercontainer), although these have not so far been addressed in KBS-3H safety studies (see also Section 11.2.2).

Scoping calculations reported in Appendix B.7 of the Evolution Report (see also Figure 6-4 of the present report) indicate that corrosion is unlikely to result in a canister lifetime of less than several hundred thousand years, even in cases where mass transport in the buffer is severely perturbed (and for a moderately pessimistic assumption regarding the groundwater flow rate at the buffer/rock interface, see Section 6.2.6). It is acknowledged that there are several simplifying assumptions in these scoping calculations that could result in somewhat higher or lower canister lifetimes than those calculated, including the effect of methane and hydrogen naturally present in the groundwater on the microbial reduction of groundwater sulphate to sulphide (the impact of repository generated hydrogen has been shown to be small). Nevertheless, a canister lifetime in excess of 100,000 years is expected, even in the event of a significantly perturbed buffer/rock interface.

Penetration of dilute water to repository depth in association with glacial retreat cannot be ruled out during future glacial cycles. As mentioned in Section 7.2.6, ongoing site characterisation work is exploring whether such events could have happened in the past. The timing of the formation of permafrost and ice sheets and of the next glacial retreat may be affected by anthropogenic emissions, especially greenhouse gases, and is therefore subject to uncertainty. However, assuming a repetition of the last glacial cycle (from the Eemian interglacial to the end of the Weichselian glaciation), the next glacial retreat and hence the next possibility for penetration of glacial meltwater to repository depth is in about 70,000 years time (see, e.g. Figure 7-1 in the Evolution Report).

Even following partial erosion of the buffer, the rate of corrosion is expected to remain low due to the limited supply of sulphide from the groundwater. The scoping calculations reported in Appendix B.7 of the Evolution Report indicate that about a million years would be required for failure by corrosion even if the entire buffer is treated as a mixing tank, based on the currently observed maximum sulphide concentration in the groundwater at Olkiluoto, and the current hydraulic gradient[20].

[20] Higher flows are expected in association with glacial retreat, but these would only persist for a few thousand years.

Radionuclide release and transport calculations addressing canister failure due to corrosion assume that failure of a single canister occurs 100,000 years in the future (Section 9.6.2). This failure time is regarded as pessimistic, given the slow rate of canister corrosion following even significant erosion of the buffer. However, in neither of the scenarios leading to failure by corrosion can an estimate currently be made of the likelihood or rate of canister failure by corrosion in a million year time frame, given the limited quantitative understanding of relevant processes, such as chemical erosion of the buffer and the impact of methane and hydrogen on the microbial reduction of groundwater sulphate to sulphide. It should be noted that, although only a single canister failure is considered, loss of buffer around one canister due to exposure to glacial meltwater may affect the corrosion rate of neighbouring canisters, since the the buffer density along the drift will tend to homogenise over time. This also means, however, that the impact on buffer density and on the corrosion rate of the first canister will diminish with time. This is in contrast to the case of KBS-3V, where buffer loss around one canister will not affect the state of the buffer around the other canisters.

8.6 Scenarios involving canister rupture due to rock shear

A single scenario is depicted in Figure 8-1 involving, as the first mode of canister failure, rupture due to rock shear, where the shear movement is in excess of 0.1 m. Significant shear movements on fractures intersecting the repository drifts are most likely to occur in association with large earthquakes. Scoping calculations reported in Appendix B.5 of the Evolution Report give the expectation value of the number of canisters in the repository that could potentially be damaged by rock shear in the event of a large earthquake as 16 out of the total number of 3,000 canisters[21], although there are some significant uncertainties associated with these values that could lead to them giving either an underestimate or an overestimate of the actual likelihood of damage (see Section 7.4.5 in the Evolution Report, /Smith et al. 2007b/). Application of a canister position avoidance criterion would potentially decrease this number, but an efficient criterion has not been developed and tested for KBS-3H (Section 11.4.4). The probability of an earthquake occurring that is sufficiently large to cause such damage in a 100,000 year time frame has been estimated as 0.02 (Table 5-8 in /La Pointe and Hermanson 2002/).

Any future large earthquakes occurring at the Olkiluoto site are not expected to be uniformly distributed in time. Palaeoseismicity studies support the suggestion that major seismic activity was, in the past, limited to a short period after the last deglaciation, and it may be inferred that this will also be the case in the future (Section 2.2.2 of the Evolution Report, /Smith et al. 2007b/). Given that, based on a repetition of the last glacial cycle, the next glacial retreat will be in 70,000 years time (see the discussion of canister failure by corrosion in Section 8.5), radionuclide release and transport calculations addressing canister failure due to rock shear also assume that failure occurs 70,000 years in the future (Section 9.6.3).

[21] For a KBS-3V repository at Olkiluoto, a higher expectation value of 20 is calculated, the difference being largely due to greater vertical extent of a KBS-3V repository and hence its greater vulnerability to movement on the relatively dense population of sub-horizontal fractures. Application of a canister position avoidance criterion, such as the Expanded Full Perimeter Criterion (EFPC) applied in SR-Can, would significantly decrease this number.

9 Radionuclide release and transport processes and analyses

9.1 Purpose and scope of radionuclide release and transport analyses

The discussions in earlier chapters indicate that, while a prolonged period of isolation of the spent fuel and containment of radionuclides in the copper canisters, as in the Base Scenario, is the expected course of evolution for a KBS-3H repository, there are evolutionary paths or scenarios that cannot currently be excluded in which one or more canisters fail, giving rise to some radionuclide releases. According to the safety concept (Figure 4-4), safety in these scenarios rests principally on complete containment of radionuclides by the remaining canisters and, for the failed canisters, slow release from the spent fuel, slow diffusive transport in the buffer, and slow transport in the geosphere to the biosphere. Each of these is, however, subject to uncertainties. Radionuclide release and transport calculations are therefore carried out to assess the robustness of the safety concept in view of these uncertainties.

Some uncertainties are treated using models, computer codes and parameter values that are conservative, meaning that they tend to over-estimate radiological consequences. Identifying what is a conservative model approach, assumption or parameter value is not, however, always straightforward – what is conservative with respect to one process may not be conservative with respect to another competing process. Furthermore, a purely conservative approach does not give a basis for deciding which uncertainties are the most important in terms of system performance. Thus, many uncertainties are treated by defining a range of assessment cases – i.e. specific model realisations of different possibilities or illustrations of how a system might evolve and perform in the event of canister failure – and analysing these cases in terms of hazard to humans and to other biota, quantified in terms of annual dose to the most exposed individuals or radionuclide-specific fluxes across the geosphere/biosphere interface (in order to compare with Finnish regulatory guidelines).

As mentioned in Section 4.1, although two broad realisations of KBS-3H design variants are currently being developed in parallel – the Basic Design and the DAWE design variant – the final saturated state of the repository is essentially the same whichever option is implemented. Differences between the design options regarding the evolution of the repository principally affect the early evolution phase, prior to any possible release of any radionuclides. Thus, the calculations reported in the present chapter are applicable to either design.

Consistent with Finnish regulations, assessment cases address radionuclide release and transport over a million year time frame. The consequences of the ultimate failure of the repository multi-barrier system in the farthest future (beyond a million years), including the possible exhumation of the repository, are briefly discussed in Chapter 10 and in the Complementary Evaluations of Safety Report.

9.2 Release and transport processes included in the analyses

9.2.1 Near-field release and transport processes

Once a canister has failed by one of the failure modes described in Chapter 8 and water enters its interior, three broad radionuclide release mechanisms may occur:

1. the cladding will slowly corrode on contact with water, resulting in the dissolution of activation products embedded within it,

2. radionuclides that have been segregated to grain boundaries in the fuel, to pellet cracks and to the fuel/sheath gap will rapidly dissolve in water (although some may also form volatile products such as methane or carbon dioxide that mix with hydrogen gas produced principally by corrosion of the insert),
3. water will start to interact with the surfaces of the fuel matrix, resulting in the release of radionuclides embedded within it, and of radioactive gases present in fission gas bubbles.

These mechanisms, already mentioned in Section 6.5 in the context of a canister with an initial penetrating defect, are described in detail in Chapter 8 of the Evolution Report.

Released radionuclides will be transported, primarily by aqueous diffusion, along continuous water pathways from the canister interior through the buffer and to the host rock. Transport will be retarded by sorption on solid surfaces and aqueous concentrations will be limited by solubility constraints.

9.2.2 Geosphere transport processes

Radionuclides will be transferred by diffusion from the buffer to flowing water in transmissive fractures intersecting the repository drift. Radionuclides will diffuse away from the buffer/rock interface and will be transported downstream by the flowing water.

Radionuclide transport in the geosphere will predominantly be by advection in channels within transmissive fractures. Diffusion will take place throughout the connected pore and fracture space in the geosphere; diffusion into stagnant pore water and sorption on fracture and rock matrix pore surfaces retards advection in fractures.

9.2.3 Repository-generated gas

Gas generated inside a canister, principally hydrogen from the corrosion of the insert subsequent to canister failure, may affect radionuclide release and transport in two broad ways. Firstly, there may be mixing of radionuclides present as volatile species (principally C-14) with repository-generated gas and subsequent migration of this gas along pathways from the canister interior through the buffer to the host rock. Secondly, the build-up of high gas pressure inside the canister following water ingress may lead to the expulsion of water and dissolved radionuclides from inside the canister into the buffer (Section 7.1.7). Although the impact of repository-generated gas is assumed to be negligible in the majority of assessment cases, some specific variant cases to evaluate its impact (cases PD-EXPELL, PD-VOL-1 and PD-VOL-2, see Table 9-1) have also been considered.

9.2.4 Biosphere processes

Radionuclides will be released from the geosphere to the biosphere either in solution in groundwater or in gaseous form. Radionuclides that are dissolved in groundwater will be transferred to surface water bodies (brooks, rivers and lakes) either directly or indirectly through the porous overburden by advection and diffusion. Some radionuclides entering the overburden will become associated with geological materials and soils, erosion of which provides a further transport mechanism, especially in agricultural lands. Furthermore, the groundwater table, changes in its level over time, and capillary rise may result in some radionuclides becoming available for the root uptake by plants, and subsequent further transfer in the food webs.

Any radioactive gases not dissolved in groundwater are expected to be ventilated relatively quickly to the atmosphere. Some, however, may be assimilated by biological organisms by inhalation, plant gas exchange and microbial metabolism and some may be captured and retained in pore spaces in the overburden.

In the surface water bodies, the radionuclide concentration is affected by water turnover and sedimentation, resuspension and erosion processes, in addition to the biological turnover.

Some radionuclides will be transferred by flowing water to the Baltic Sea and further to the oceans, while others will remain in the Olkiluoto area.

Microbial processes are common in all parts of the biosphere. In biosphere modelling, they are usually included either in the sorption or the biological parameters, depending on the specific process under consideration and on the availability of data.

9.3 Assessment cases

The definition of an assessment case includes (i), the canister failure mode that is presumed to lead to the formation of transport pathways between the canister interior and its surroundings, and (ii), the models and data that describe subsequent radionuclide release and transport. The approach adopted to select assessment cases involves the following main steps:

- identify plausible sequences of events or processes (scenarios) potentially leading to canister failure within a million year time frame, and the canister failure modes to which these scenarios give rise – scenarios and failure modes have already been identified in Chapter 8,
- for each canister failure mode, define a Base Case against which to compare the results of variant cases,
- define a number of variant cases that illustrate the impact of specific uncertainties on the radiological consequences of canister failure.

The grouping of assessment cases according to the canister failure mode that they address is illustrated in Figure 9-1.

The models and data that define each case are individually specified according to the judgment of the safety assessor. This "deterministic" approach has been adopted, rather than a "probabilistic" approach in which parameter values are sampled randomly from probability density functions (PDFs). A deterministic approach can give a clear illustration of the impact of specific uncertainties. Furthermore, it avoids the need to define PDFs that quantify in single distributions widely different types of uncertainty (e.g. "aleatory" uncertainties related to variability or randomness and "epistemic" uncertainties arising, for example, where there is a range of plausible alternative models consistent with current scientific knowledge). Furthermore, in the present safety assessment, the treatment of some uncertainties involves model assumptions that are hypothetical and highly conservative (e.g. the treatment of a perturbed buffer/rock interface as a highly conductive "mixing tank"), and it is unclear whether or not it is meaningful to assign a probability attached to such assumptions. However, both deterministic and probabilistic approaches have advantages and disadvantages, as discussed in Section 5.7.2 of the Complementary Evaluations of Safety Report, and, in some recent safety assessments, a combination of the two approaches has been employed. Examples of combined approaches are SR-Can /SKB 2006a/ and the Swiss Project Opalinus Clay /Nagra 2002/.

The selection and detailed definition of assessment cases to evaluate radionuclide release and transport is described in the Radionuclide Transport Report /Smith et al. 2007a/. The process is, in general, more subjective than scenario identification. Parameters in the Base Cases are, in most instances, selected to be either realistic or moderately conservative in the sense that they are expected to lead to an overestimate of radiological consequences. Furthermore, perturbations to radionuclide release and transport caused, for example, by the steel and cementitious components of the KBS-3H repository external to the canisters are assumed to be negligible in the Base Cases (even though this may be non-conservative), but are considered in variant cases. The variant cases are chosen to cover the various scenarios that have been identified as leading to loss or major degradation of repository safety functions, and to canister failure. In addition to these "scenario uncertainties", there are additional uncertainties that have a more limited impact on the repository safety functions. The variant cases for the most part take a more pessimistic view of uncertainties than the Base Cases.

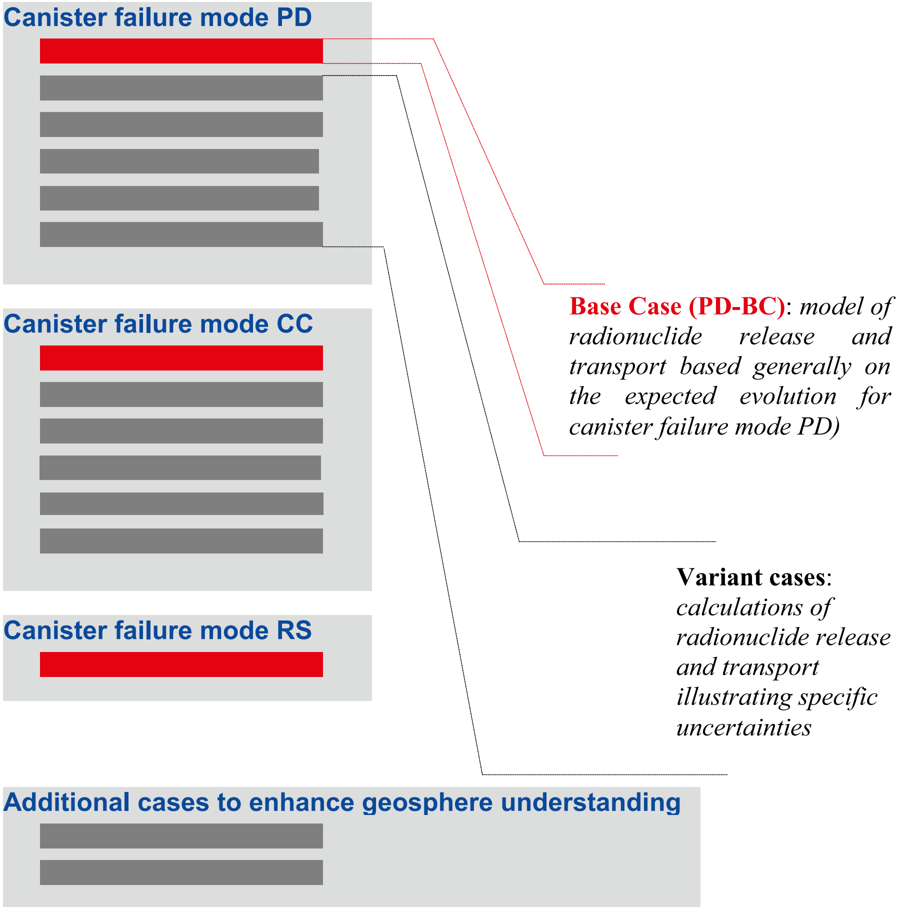


Figure 9-1. Grouping of assessment cases according to the canister failure mode postulated. PD: initial, penetrating defect; CC: failure due to corrosion of the copper canister shell; RS: rupture due to rock shear and the transfer of shear stresses from the rock via the buffer to the canister (in particular, in the event of post-glacial earthquake).

Process tables that summarise the handling of internal processes in the safety assessment are used as check lists to ensure that no important processes and associated uncertainties have been overlooked in the identification of scenarios and assessment cases. Process tables for a KBS-3H repository at Olkiluoto are given in the Process Report. Furthermore, a detailed comparison of the calculational cases with those of TILA-99 and SR-Can has confirmed that there are no omissions or gaps in the KBS-3H assessment apart from where limitations related to the scope of the assessment mean that the treatment of some uncertainties is put aside at this stage (see Complementary Evaluations of Safety Report).

An overview of assessment cases is given in Table 9-1. Each case is assigned a unique name, comprising two parts separated by a hyphen. The first part of the name indicates the canister failure mode that the case addresses:

- PD: canister with an initial penetrating defect,
- CC: canister failure due to copper corrosion,
- RS: canister failure due to rock shear.

The second part of the name identifies the case either as a Base Case (BC) for a given canister failure mode, or a variant case illustrating the impact of one or more uncertainties.

Cases MD-1, MD-2 and MD-3 illustrate the impact of rock matrix diffusion depth for a hypothetical pulse release to the geosphere, and therefore are not associated with a specific canister failure mode.

Table 9-1. Overview of assessment cases.

Cases assuming a single canister with an initial penetrating defect (PD-)	
Case	**Description**
PD-BC	Base Case for initial penetrating defect in BWR-type canister
PD-VVER	Initial penetrating defect in VVER-440 PWR type canister
PD-EPR	Initial penetrating defect in EPR type canister
PD-HIFDR	Increased fuel dissolution rate
PD-LOFDR	Reduced fuel dissolution rate
PD-IRF[a]	Evaluates transport only of radionuclides present in instant release fraction (see footnote a below this table)
PD-BIGHOLE	Increased defect size
PD-HIDELAY	Increased delay until loss of defect transport resistance
PD-LODELAY	Decreased delay until loss of defect transport resistance
PD-BHLD	Increased defect size plus decreased delay until loss of defect transport resistance
PD-HIDIFF	Increased diffusion rate in buffer
PD-FEBENT1	Perturbed buffer-rock interface – high conductivity, narrow perturbed zone
PD-FEBENT2 PD-FEBENT3	Perturbed buffer-rock interface – more extensive perturbed zone (2 different thicknesses)
PD-SPALL	Perturbed buffer-rock interface – high conductivity, narrow perturbed zone, lower flow through intersecting fractures than that assumed in cases PD-FEBENT1, 2 and 3
PD-EXPELL	Dissolved radionuclides expelled by gas from canister interior and across buffer to geosphere
PD-VOL-1 PD-VOL-2	C-14 transported in volatile form by gas generated by corrosion (2 rates of gas generation)
PD-BCN	Initial penetrating defect in BWR-type canister; Nb present in near field and geosphere in anionic form
PD-BCC	Initial penetrating defect in BWR-type canister; C-14 present in geosphere in anionic form (carbonate)
PD-VVERC	Initial penetrating defect in VVER-440 PWR type canister; C-14 present in geosphere in anionic form (carbonate)
PD-EPRC	Initial penetrating defect in EPR type canister; C-14 present in geosphere in anionic form (carbonate)
PD-NFSLV	Near-field solubilities varied according to uncertainties in redox conditions
PD-SAL	Brackish/saline water present at repository depth (all time)
PD-HISAL	Saline water present at repository depth (all time)
PD-GMW PD-GMWV	Change from reference (dilute/brackish) water to glacial meltwater at 70,000 years (release also starts at 70,000 years – two alternative meltwater compositions)
PD-GMWC	Change from reference (dilute/brackish) water to glacial meltwater at 70,000 years (release starts at 1,000 years, as in the reference case)
PD-HIFLOW	Increased flow at buffer-rock interface
PD-LOGEOR	Reduced geosphere transport resistance
PD-HIGEOR	Increased geosphere transport resistance
PD-HIFLOWR	Increased flow at buffer-rock interface and reduced geosphere transport resistance

Table 9-1. (Continued) Overview of assessment cases.

Cases assuming a single canister failing due to copper corrosion (CC-)	
Case	**Description**
CC-BC	Base Case for failure due to copper corrosion; buffer treated as mixing tank
CC-HIFDR	Increased fuel dissolution rate
CC-LOFDR	Reduced fuel dissolution rate
CC-GMW	Glacial meltwater present at repository depth (impact on near-field solubilities and geosphere retention parameters)
CC-LOGEOR	Reduced geosphere transport resistance
CC-LOGEORG	Reduced geosphere transport resistance, glacial meltwater[b]
CC-LOGEORS	Reduced geosphere transport resistance, saline groundwater[b]
Cases assuming a single canister failing due to rock shear (RS-)	
Case	**Description**
RS-BC	Base case for failure due to rock shear
RS-GMW	Glacial meltwater present at repository depth (impact on near-field solubilities and geosphere retention parameters)
Additional cases (hypothetical pulse release to geosphere) (MD-)	
Case	**Description**
MD-1	Variations in matrix diffusion depth (3 cases)
MD-2	
MD-3	

a Certain radionuclides are enriched at grain boundaries in the fuel, at pellet cracks and in the fuel/sheath gap as a result of thermally driven segregation during irradiation of the fuel in the reactor. These are assumed to enter solution rapidly once water contacts the fuel pellet surfaces, and are termed the instant release fraction (IRF).

b Glacial meltwater is a very dilute ice-melting water. Saline groundwater represents water with a Total Dissolved Solid (TDS) content of about 20 g/l. For detailed composition of the waters used in the assessment, see Appendix D of Radionuclide Transport report.

Given that a key question addressed by the KBS-3H safety studies is whether or not there are safety issues identified in the KBS-3V/KBS-3H difference analysis with the potential to lead to unacceptable radiological consequences, a number of assessment cases are defined addressing uncertainties related to features and processes that have a different significance to, or potential impact on, KBS-3H compared with KBS-3V. These cases, which are defined in Table 9-1, have already been mentioned in Chapter 7 in the discussion of key safety issues with different significance to, or potential impact on, KBS-3H compared with KBS-3V. They are:

- PD-HIDIFF,
- PD-FEBENT1, PD-FEBENT2, and PD-FEBENT3,
- PD-SPALL,
- PD-EXPELL.

A number of additional cases are also analysed to illustrate the impact of other uncertainties in key features that are common to the KBS-3H and KBS-3V safety concepts.

The majority of variant cases are defined for the initial penetrating defect canister failure mode. Using the initial penetrating defect as a reference failure mode for exploring uncertainties provides a common basis for comparison with the earlier Finnish TILA-99 safety assessment /Vieno and Nordman 1999/, and is also the approach used in SR-Can. It should be noted that many of these uncertainties are not specific to any particular canister failure mode. The assumption of an initial penetrating defect results in the earliest possible radiological impact, although not necessarily the largest impact, for each uncertainty considered.

9.4 Models, datasets and computer codes

9.4.1 General approach to model assumptions and parameter selection

In defining the Base Cases, combinations of multiple, highly conservative assumptions are avoided where these are judged to be implausible and potentially to lead to unrealistically high radiological consequences. In general, either realistic or moderately conservative model assumptions and parameter values are generally selected in the Base Cases, with variant cases assigned either more optimistic or (more often) more conservative assumptions and values, within the identified ranges of uncertainty.

The variant cases are designed to illustrate specific uncertainties or combinations of uncertainties, including (but not restricted to) those that are of particular relevance to KBS-3H. However, not all conceivable uncertainties and combinations of uncertainties are covered. For example, uncertainties in the transport barrier provided by the geosphere, biosphere uncertainties and uncertainties related to future human actions are either not addressed or are analysed in less detail than others.

In some cases, whether one alternative parameter value is more optimistic or conservative than another is well known from experience in past assessments or is clear from the nature of the release and transport processes involved. For example, a high geosphere transport resistance is clearly a more optimistic assumption than a low transport resistance. On the other hand, in other cases, a sensitivity analysis would, in principle be necessary to explore the impact of variations in one or more parameters on radionuclide releases, before it could be stated what is an optimistic parameter value, and what is conservative. Such sensitivity analyses have been performed for the transport models in the biosphere /Broed 2007/ but not, in general, for other models in the present assessment.

A comprehensive data report, which would include, for example, sensitivity analyses to justify the conservatism of specific parameter values, will be considered in any future safety studies of the KBS-3H repository, and is likely to be required in support of a future safety case.

9.4.2 The near field

For modelling near-field release and transport in the Base Cases and the variant cases, extensive use has been made of SR-Can parameter values and model assumptions, except where these are affected by differences in the materials to be disposed of in Finnish and Swedish repositories, differences in the geometry of the engineered barriers and differences between conditions at Olkiluoto and those at the Swedish sites considered in SR-Can. Where differences arise, the selection of parameter values and model assumptions has been made largely according to "expert judgement" (see, for example, Appendix E of the Radionuclide Transport Report) based on considerations such as use in previous assessments, additional data gathering and laboratory studies.

Near-field analyses have been performed with the REPCOM code. REPCOM has been developed by the Technical Research Centre of Finland (VTT) for radionuclide transport analyses in the near field of repositories for low- and intermediate-level waste and spent fuel. The code is described further in Appendix A of the Radionuclide Transport Report, including verification aspects. In addition, near-field releases in case PD-BC calculated using REPCOM have been compared with those calculated using an alternative near-field code: the SPENT code used by Nagra in recent safety assessments in Switzerland in Appendix B of the Radionuclide Transport Report. The, mostly small, differences in the results obtained using the two codes are explained in terms of differences in underlying model assumptions.

9.4.3 The geosphere

In reality, geosphere transport takes place in a network of fractures with significant variability in their flow and transport properties. The highly simplified geosphere transport modelling carried out in the present safety assessment, however, considers a single, representative geosphere fracture that intersects a deposition drift near to the location of a failed canister (see Section 9.6.1).

The modelling approach and parameter values used are based largely on TILA-99, although more recent developments in the understanding of the Olkiluoto site, and, in particular, discrete fracture network modelling carried out in support of the KBS-3H safety studies /Lanyon and Marschall 2006/, are used to provide additional support for the parameter values selected (for example, in terms of their conservatism).

Geosphere (far-field) analyses have been performed with the FTRANS code /FTRANS 1983, Nordman and Vieno 1994/. The code is again described further in Appendix A of the Radionuclide Transport Report, including verification aspects.

No attempt has been made to model the entire fracture network. Rather, in each assessment case, radionuclide releases from the buffer around a single failed canister are transferred to the single, representative geosphere fracture mentioned above, assuming specified flow conditions at the buffer/rock interface. Uncertainty and variability in the geosphere are considered by defining and analysing variant assessment cases with, for example, reduced or increased geosphere transport resistance (see Table 9-1). These analyses are limited, since uncertainties related to the geosphere transport barrier do not fall within the main focus of the present safety assessment. However, a detailed analysis of the current understanding of geosphere transport parameters is being undertaken in the context of the safety case for a KBS-3V repository at Olkiluoto (Section 11.1).

9.4.4 The biosphere and the evaluation of annual landscape dose

Biosphere modelling is divided into (i), predictions of the physical terrain and the ecosystems possibly receiving contaminant releases from the repository (referred to as “forecasts”, below), and (ii), modelling the transport of radionuclides in the biosphere. The forecasts are produced by the terrain and ecosystems development model (TESM; /Ikonen et al. 2007/). Spatial distributions of radionuclide concentrations in the biosphere are the output from the landscape model (LSM; /Broed 2007/).

The climatic variations assumed by the TESM in making the forecasts are based on a repetition of the last glacial cycle (from the Eemian interglacial to the end of the Weichselian glaciation). Lakes, rivers and their catchment areas are identified using standard GIS (geographical information system) processing tools, specifically the approach found suitable for the site by / Ojala et al. 2006/ and further adjusted in /Ikonen et al. 2007/. The model of /Brydsten 2004/ was applied for simulating the accumulation of sediments and reed growth in the lakes and coastal areas, with rate functions updated for the site (for details, see /Ikonen et al. 2007/). On the basis of the TESM, biosphere objects (forests, wetlands, lakes, rivers and coastal areas possibly receiving even indirectly any contamination from the repository) were identified at each time step of the forecast and their geometrical properties calculated.

The LSM is a time-dependent linked transport model containing the above-mentioned biosphere objects represented by ecosystem-specific compartment models /Broed 2007/, which are mostly the same as in SR-Can. The connections between the objects were derived from the terrain forecasts for the period from the present to 8,000 years in the future with 500-year intervals. The LSM is implemented in Pandora /Åstrand et al. 2005/, which is a tool developed by Facilia AB and used by SKB and Posiva for biosphere modelling. Pandora is based on the Matlab/ Simulink© environment (www.mathworks.com).

The LSM can currently not be applied to releases of gaseous C-14 from the repository. A set of simplified models, based on a specific activity approach, has been developed for assessment of human exposures resulting from potential underground releases of C-14 /Avila and Pröhl 2007/.

In the safety assessment, the annual effective dose to the most exposed individual calculated using the above models is termed the annual landscape dose. It is the primary dose endpoint for the "environmentally predictable future" (see below). It order to calculate this indicator, the most exposed individual is assumed to spend all his or her time within the single biosphere object producing the highest dose, including external exposure as well as internal exposure from eating and drinking food produced in and water available from that same biosphere object /Avila and Bergström 2006, Broed 2007, Broed et al. 2007/.

9.5 Assessment endpoints

Finnish regulations, as summarised in Chapter 2, distinguish between the "environmenttally predictable future" (assumed to last about ten thousand years), during which conservative estimates of doses must be made, and the era of "large-scale climate changes" (beyond about ten thousand years) when periods of permafrost and glaciations are expected, and radiation protection criteria are based on constraints on nuclide-specific activity fluxes from the geosphere ("geo-bio flux constraints").

The primary endpoints used in the present safety assessment are:

- annual landscape dose, calculated over the "environmentally predictable future" using the biosphere modelling approach summarised in Section 9.4.4, above, and
- geo-bio fluxes, which are calculated up to a million years, and are used for comparison with Finnish regulatory geo-bio flux constraints (Table 2-1) at times beyond the "environmentally predictable future".

In addition, a safety indicator based on an indicative stylised well scenario – WELL-2007 dose – has been calculated for all assessment cases. Ingestion of contaminated water by humans is the only exposure pathway considered in this stylised well scenario[22]. WELL-2007 dose refers to committed effective[23] doses due to ingestion of water over one year, where the effects of ingestion are integrated over the adult life of an individual human /ICRP 1991/. Dose conversion factors for WELL-2007 are given in the Radionuclide Transport Report. Calculation of WELL-2007 dose further facilitates comparison with regulatory guidelines for the "environmentally predictable future", as well as the results from other safety assessments and safety cases, without the need to justify a wide range of biosphere modelling assumptions. Given the simplified nature of WELL-2007, however, comparison of WELL-2007 dose with regulatory constraints is not regarded by itself as an adequate test of compliance with the regulatory guidelines, which is why the annual landscape dose is also calculated.

Exposure of other (non-human) biota is not explicitly addressed in the present safety assessment, but is considered in the Biosphere Analysis Report /Broed et al. 2007/.

Geo-bio fluxes and WELL-2007 doses have been calculated for all assessment cases. Results are presented in Sections 9.6 (Base Case analyses) and 9.7 (variant cases addressing issues with different significance to, or potential impact on, KBS-3H compared with KBS-3V). Annual landscape dose – the primary quantity used to assess compliance with Finnish regulations in the environmentally predictable future – has been calculated for a more limited set of cases, namely those in which radionuclide release begins within the first ten thousand years. The results are presented in the discussion of compliance in Section 9.9.4.

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[22] A second stylised well scenario that includes additional exposure pathways, AgriWELL-2007, has also been used in calculations described in the Biosphere Analysis Report /Broed et al. 2007/.

[23] Effective dose is used in radiological protection to relate exposure, internal or external, to ionising radiation to stochastic effects, such as the induction of cancer and hereditary effects.

9.6 Base Case analyses

9.6.1 Canister with an initial penetrating defect

a. General assumptions

The discussion in Section 6.5 (which summarises a more detailed discussion in Chapter 8 of the Evolution Report) illustrates that there are considerable uncertainties regarding the internal evolution of a canister with an initial penetrating defect. The defect will have a relatively minor impact on the containment function of the canister as long as it remains small or becomes rapidly plugged with bentonite and iron corrosion products, limiting the supply of water to the surface of the insert and hence its rate of corrosion. On the other hand, the possibility that the corrosion products will lead to an expansion of the defect cannot currently be excluded. This could in turn lead to further corrosion, weakening and eventual failure of the insert, which would allow the water to contact the fuel and cladding and radionuclides to dissolve and migrate out of the canister. It is this possibility that is pessimistically assumed in the Base Case and variant assessment cases dealing with this canister failure mode.

In the Base Case (case PD-BC), the initial penetrating defect is assumed to affect a single canister of BWR fuel from the Olkiluoto 1–2 reactors. Steady groundwater flow and geochemical conditions are assumed at all times. Groundwater conditions are assumed to be reducing and dilute/brackish. Of the various groundwaters studied (Appendix D of the Radionuclide Transport Report), dilute/brackish groundwater is closest in terms of total dissolved solids (TDS) to the expected undisturbed conditions at repository depth in the period up to 10,000 years in the future /Pastina and Hellä 2006/.

The geometry of the domain represented by the Base Case near-field model is illustrated in Figure 9-2.

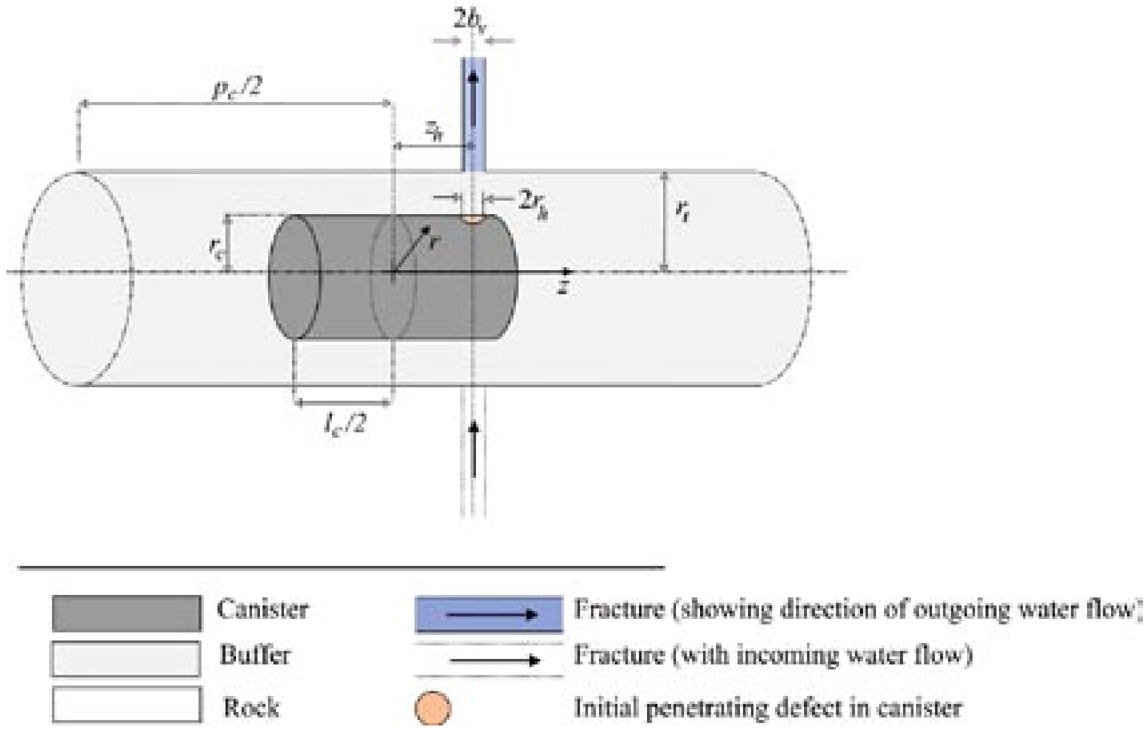

Figure 9-2. *Geometrical domain of the near field model in the Base Case for an initial penetrating defect. The canister has a length (l_c) of 4,835 m and an outer diameter ($2r_c$) of 1.05 m. The drift diameter ($2r_t$) is 1.85 m. The canister pitch (p_c) is 11 m. The defect diameter ($2r_h$) is 1 mm. Other geometrical parameters and parameter values are defined in the Radionuclide Transport Report /Smith et al. 2007a/.*

A drift section containing a canister with an initial penetrating defect is assumed to be intersected by a single fracture with a transmissivity (3×10^{-9} m^2 s^{-1}) that is judged to be moderately pessimistic, although the possibility of more transmissive fractures intersecting the drift as canister deposition locations cannot be excluded (see the discussion in Section 5.2 of the Radionuclide Transport Report). Conservatively, it is assumed that the canister is located at a position that minimises the transport distance across the buffer between the defect and the fracture mouth (i.e. the centre plane of the fracture is assumed to pass through the centre of the defect, as illustrated in Figure 9-1). The diameter of the initial penetrating defect is taken to be 1 mm. This corresponds roughly to the maximum defect size that might escape detection using current non-destructive testing (NDT) quality control techniques (a larger, 4 mm defect is, however, considered in the variant cases PD-BIGHOLE and PD-BHLD, see Table 9-1).

Following SR-Can, it is assumed to take 1,000 years for water to contact the fuel, the Zircaloy and other metal parts and for a transport pathway to be established between the canister interior and exterior. This assumption is based on the slow water ingress rate, further decreased by the gradual build-up of an internal counter pressure due to hydrogen gas formation, as well as on the barrier functions of the cast iron insert and of the fuel cladding. According to Section 10.5.2 of /SKB 2006a/, 1,000 years can be regarded as pessimistic, since any one of these factors is likely to provide more than 1,000 years of delay. Thereafter, the supply of water to the canister interior is conservatively assumed to be unlimited, but the defect provides a continuing transport resistance for released radionuclides. After the ingress of water, fuel dissolution is assumed to take place at a constant fractional rate of 10^{-7} per year, with congruent release of radionuclides. This represents the peak of a triangular distribution recommended for use in SR-Can by /Werme et al. 2004/. It is assumed that the inventory of activation products in Zircaloy and other metal parts is released congruently with the corrosion of the metal (a more pessimistic approach is taken in SR-Can, where no credit is taken for the delay due to the limited rate of metal corrosion).

Solubility limits, sorption coefficients and other transport parameters for the near field and geosphere are described in the Radionuclide Transport Report. As was the case in TILA-99, the solubility limits are applied in the near field model only inside the canister and at the buffer/rock interface, but not throughout the buffer. The effects of this model simplification are evaluated in additional calculations using the Nagra near-field code SPENT (Appendix B of the Radionuclide Transport Report).

Anion exclusion is treated by assigning the buffer a lower porosity and a lower effective diffusion coefficient when modelling anion transport compared with the values for neutral and cationic species, the porosity assumed for neutral and cationic species being equal to the actual porosity of the buffer.

Following the stylised analysis of the "growing pinhole failure mode" in SR-Can (Section 10.5 in /SKB 2006a/), it is assumed that the defect ceases to provide transport resistance at 10,000 years after canister deposition (9,000 years after radionuclide transport pathways from the canister interior are established). It should be noted, however, that the SR-Can Data Report (Section 4.4.7 of /SKB 2006b/) suggests that loss of transport resistance could occur at any time between 1,000 and 100,000 years after radionuclide transport pathways from the canister interior are established, and the choice of 10,000 years as the Base Case parameter value is somewhat arbitrary. Furthermore, loss of transport resistance may be a process that occurs gradually over time, rather than as a discrete event. An instantaneous loss of transport resistance is, however, a conservative assumption, since a gradual loss of transport resistance would spread the peak release over a longer period of time, reducing its magnitude.

Geosphere transport parameter values are taken for the most part from TILA-99, using data for reducing and, where relevant, non-saline conditions representing fresh groundwater conditions in the rock (TDS < 1 g/l). The TILA-99 value of 50,000 years m^{-1} assigned to the geosphere transport resistance parameter WL/Q was based, in the first place, on statistical data for hydraulic conditions at the Olkiluoto site /Löfman 1996/. The conservatism of this choice is, however, supported by the more recent discrete fracture network modelling of the Olkiluoto site carried out by /Lanyon and Marschall 2006/, as discussed in Section 5.2 of the Radionuclide Transport Report.

b. Calculational results

Figure 9-3 shows the WELL-2007 dose (committed effective doses due to ingestion of water from a stylised well over one year, integrated over the adult life of an individual human, as described in Section 9.5) as a function of time. The figure shows total dose and the contribution to dose from individual radionuclides.

To place the timescales covered by these and subsequent figures in perspective, background shading is used to distinguish three successive time frames:

- 0 to 10,000 years.

This is Posiva's interpretation of the "environmentally predictable future". The Finnish regulatory dose constraint applies over this time frame.

- 10,000 to 50,000 years.

Assuming a repetition of the last glacial cycle (from the Eemian interglacial to the end of the Weichselian glaciation), this period will be characterised by alternating permafrost and temperate climate phases /Pastina and Hellä 2006/.

- 50,000 to 1,000,000 years.

Again assuming a repetition of the last glacial cycle, after 50,000 years the Olkiluoto area will, for the first time in the future, be covered with ice and snow. 1,000,000 years is the time at which calculations of radionuclide release and transport are terminated in the present safety assessment.

There is a sharp increase in annual landscape dose starting at about 9,000 years, associated with the loss of transport resistance of the defect. Note that this loss of transport resistance is, in fact, assumed to occur at 10,000 years in the near-field model of this cases. The earlier calculated onset the dose increase is an artefact of the finite time step size in geosphere transport modelling.

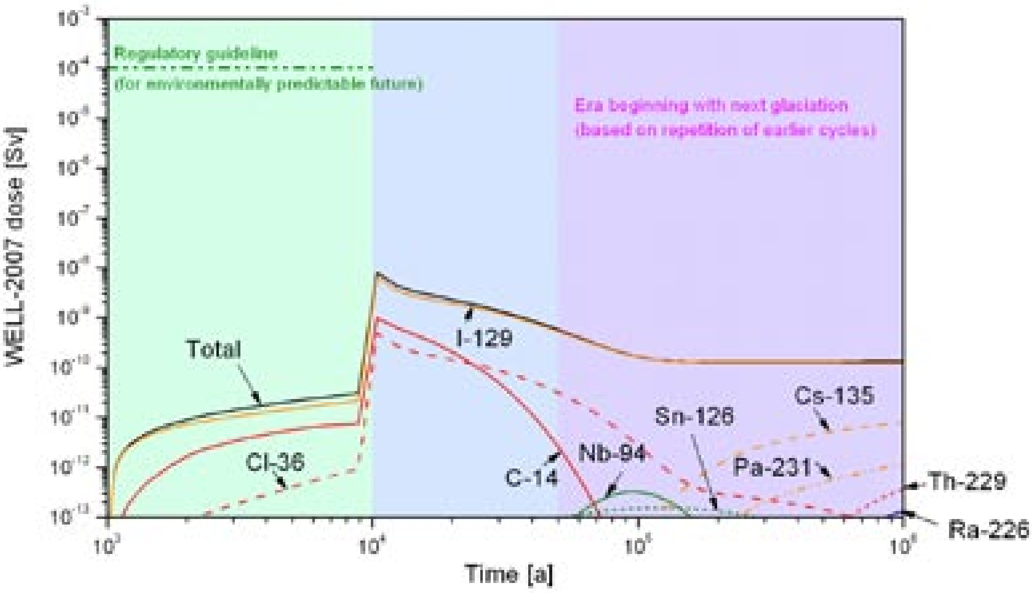

Figure 9-3. WELL-2007 dose as a function of time in case PD-BC.

The dose maximum (8×10^{-9} Sv for this case where a single canister failure is assumed) occurs shortly after loss of transport resistance of the defect. Calculated dose is at all times dominated by I-129, with other significant contributions from C-14 (up to a few tens of thousands of years, after which this radionuclide substantially decays), Cl-36 (up to about 100,000 years), and, at later times, Cs-135.

Figure 9-4 shows time-dependent releases from the geosphere to the biosphere divided by the geo-bio flux constraints specified by the Finnish regulator and given in Table 2-1. Radionuclide-specific curves are shown, as well as a curve corresponding to the sum over all calculated radionuclides.

According to the Finnish regulator STUK, the sum of the ratios of nuclide-specific activity releases to their respective constraints shall be less than one in order to satisfy regulatory requirements. From Figure 9-4, the sum of these ratios has a maximum (2×10^{-3} for this case where a single canister failure is assumed) that also occurs shortly after 10,000 years.

The results of this and other assessment cases show that, for some radionuclides, activity releases to the biosphere and the corresponding WELL-2007 doses are still increasing at the end of the million year assessment period, although in most cases[24], the magnitude of these releases, divided by the relevant geo-bio flux constraints, and doses are well below the maxima. This is the case, for example, for Cs-135 in Figures 9-3 and 9-4. The late break-through of this and other sorbing radionuclides illustrates their slowness of transport through the multi-barrier system. It is radionuclides such as I-129, Cl-36 and C-14, which are assumed to undergo little or no sorption in the buffer and geosphere that generally dominate calculated releases and doses until near the end of the assessment period. Although not calculated, Cs-135 release to the biosphere would be expected to fall shortly after a million years as its inventory becomes depleted by radionuclide decay, its half life being 2.3×10^{6} years.

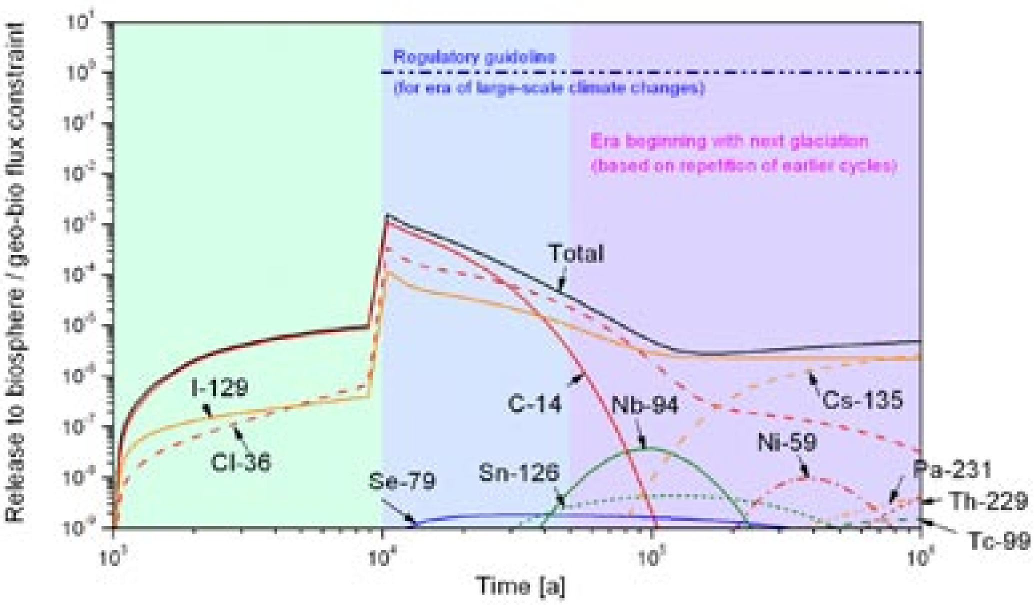

Figure 9-4. *Ratios of nuclide-specific activity releases to their respective geo-bio flux constraints in case PD-BC.*

[24] See, however, Figures 9-8 and 9-9 and the discussion thereof.

9.6.2 Canister failure due to copper corrosion

a. General assumptions

As discussed in Section 8.5, two scenarios can be envisaged in which canister failure due to copper corrosion occurs before a million years. These are scenarios in which:

1. the buffer/rock interface is perturbed, leading to enhanced mass transfer at the interface,
2. dilute glacial meltwater penetrates to repository depth, leading to chemical erosion of the buffer and to advective conditions becoming established within it.

In the Base Case and variant cases for this canister failure mode, radionuclide release is assumed to begin at 100,000 years following emplacement, and this failure time is regarded as pessimistic, given the slow rate of canister corrosion even in these scenarios.

A single representative canister, with no initial penetrating defect, is assumed to fail completely at 100,000 years. The failed canister and insert are assumed to offer no resistance to water ingress and radionuclide release. Following the ingress of water, the buffer is assumed to provide no delay or attenuation of radionuclide releases (advective conditions prevail in the buffer in the scenario where the buffer is severely eroded by glacial meltwater). Thereafter, it is further conservatively assumed that, with the exception of uranium isotopes, all radionuclides released to solution are transferred directly and instantaneously to the geosphere, with no application of near-field solubility limits. Uranium is present in relative large amounts, and may be precipitated as the fuel matrix dissolves. Although it is uncertain whether or not uranium colloids would be filtered by an eroded buffer, it is argued in SR-Can that, in terms of calculated individual doses or risks, it is conservative to assume that all precipitated uranium remains within the canister, where it continues to produce daughter radionuclides, such as the Th-230, Th-229 and Pa-231.

Following canister failure, the same steady groundwater flow and geosphere transport resistance are assumed in this Base Case as in the Base Case for a canister with an initial penetrating defect. As described in Section 9.6.1, the choice of transmissivity of the fracture intersecting the drift at the location of the failed canister and the geosphere transport resistance are considered to be moderately pessimistic. Pessimistically chosen parameters related to groundwater flow are also appropriate for this failure mode, especially because the canister positions most vulnerable to failure will be those associated with the highest groundwater flows at the buffer/rock interface. In addition to the Base Case, therefore, a number of variant cases are considered with a still more pessimistically chosen geosphere transport resistance (Table 9-1).

Geochemical conditions following canister failure in this Base Case are also the same as in the Base Cases for a canister with an initial penetrating defect. Although one scenario leading to this failure mode considers an influx of dilute glacial meltwater to repository depth, groundwater composition will vary with time in an uncertain manner following this event. Variant cases assuming both glacial meltwater and saline groundwater are therefore considered (Table 9-1).

The consequences of multiple canister failures at around the same time can, in principle, be obtained by multiplying the results for single canister failure by the number of canister failures postulated. As noted in Section 8.5, however, in neither scenario leading to this canister failure mode can an estimate currently be made of the likelihood or rate of canister failure by corrosion in a million year time frame, given the limited quantitative understanding of relevant processes, such as chemical erosion of the buffer and the impact of methane and hydrogen on the microbial reduction of groundwater sulphate to sulphide. Results are given below for a single failed canister. The possibility of multiple failures occurring within a limited time interval will, however, need to be assessed in the course of future studies.

b. Calculational results

Figure 9-5 shows the releases from the geosphere to the biosphere as a function of time in the Base Case for canister failure due to copper corrosion (case CC-BC) expressed as WELL-2007 dose. Figure 9-6 shows time-dependent releases from the geosphere to the biosphere divided by the geo-bio flux constraints specified by the Finnish regulator and given in Table 2-1, and the sum of these releases over all calculated radionuclides.

The peak annual effective dose, which is dominated by I-129, occurs shortly after canister failure at 100,000 years and takes a value of 3×10^{-7} Sv for a single canister failure. At later times, dose continues firstly to be dominated by I-129, and later by Th-229 (a decay product of Np-237). The sum of time-dependent releases from the geosphere to the biosphere divided by their respective geo-bio flux constraints also has its maximum shortly after 100,000 years, and takes a value of 6×10^{-3} (i.e. it would require in excess of one hundred canisters failing at similar times for the regulatory guideline to be exceeded; as noted earlier, the estimation of the number of canisters that may fail at similar times by this mode is an issue for future work).

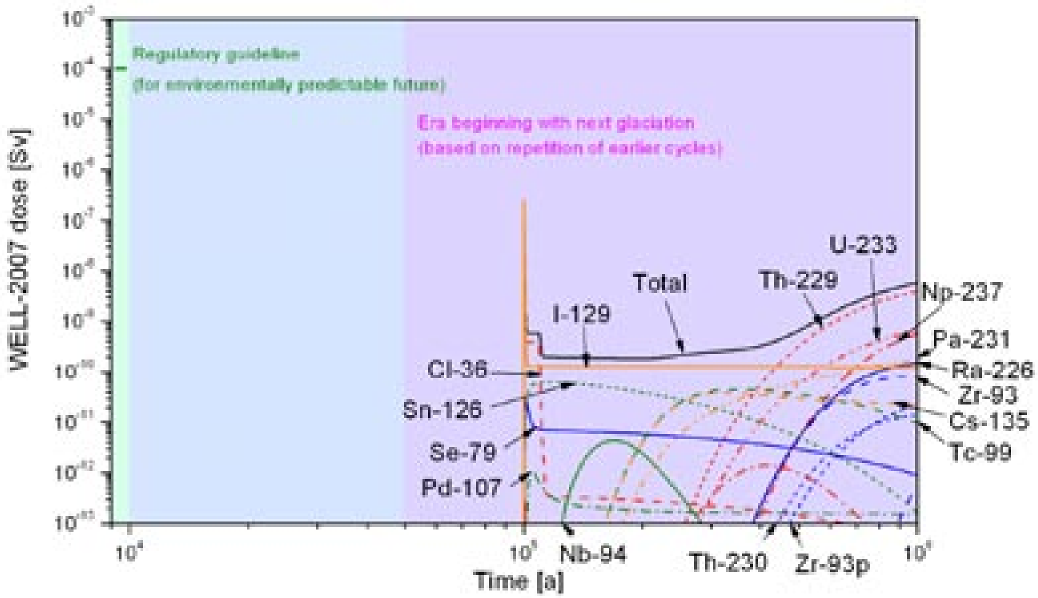

Figure 9-5. *WELL-2007 dose as a function of time in case CC-BC.*

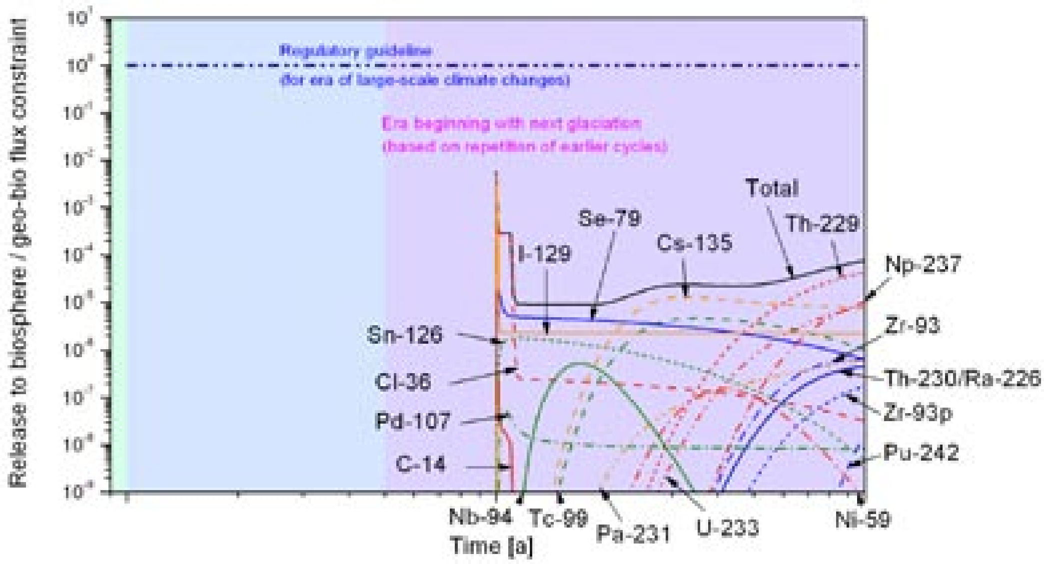

Figure 9-6. *Ratios of nuclide-specific activity releases to their respective geo-bio flux constraints in case CC-BC.*

It should be noted that the magnitude of the peak shortly after 100,000 years is increased by the conservative assumption that the canister fails completely and instantaneously at this time. In reality, some mass transport resistance may be retained following the initial failure of a canister.

9.6.3 Canister failure due to rock shear

a. General assumptions

As described in Sections 7.2.5 and 8.6, large post-glacial earthquakes in the vicinity of the repository could lead to rock movements on fractures intersecting the deposition drifts. In the Base Case and variant cases for this canister failure mode, the location of the affected canister is assumed to coincide with the location of the shearing fracture. This follows the definition of a similar case in SR-Can (Section 10.7 of SR-Can, /SKB 2006a/). Any residual resistance provided by the canister to water ingress and radionuclide release following rupture due to rock shear is not considered quantifiable and is conservatively neglected. There is also assumed to be no delay between failure and the establishment of a radionuclide pathway from the canister interior to the buffer.

Radionuclides are assumed to be released to the buffer at the inner surface of an annular cylindrical region (indicated by dashed lines in Figure 9-7) with its axis lying along that of the drift. It has an inner radius equal to $r_c + d_s$ – i.e. the canister radius plus the assumed shear displacement of 10 cm, and an outer radius equal to $r_t - d_s$ – i.e. the drift radius minus 10 cm. Note that 10 cm is a conservative estimate of the minimum shear movement required to cause canister failure; see Section 7.4.5 of the Evolution Report. A shear movement of 10 cm thus reduces the minimum transport distance for radionuclides through the buffer by 20 cm. These simplifications are made to maintain the cylindrical symmetry of the near field, which facilitates near-field transport modelling.

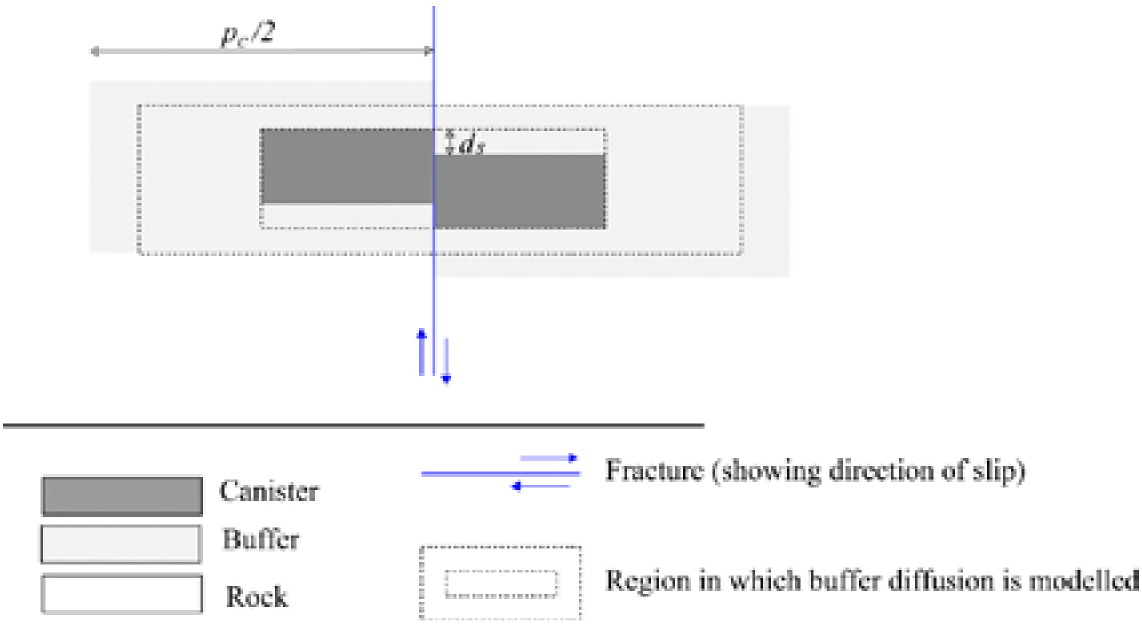

Figure 9-7. Geometrical domain of the near-field model in the Base Case for canister rupture due to rock shear. The canister has a length of 4.835 m and an outer diameter of 1.05 m. The drift diameter is 1.85 m. The canister pitch (p_c) is 11 m. The displacement due to rock shear (d_s) is 10 cm. Other geometrical parameters and parameter values are defined in the Radionuclide Transport Report /Smith et al. 2007a/.

The likely correlation between fracture size and transmissivity means that, at drift locations where rupture by rock shear occurs, the transport resistance provided by the host rock is likely to be relatively low, and may be further reduced by the effects of rock shear on the fracture. The flow in the fracture is therefore assumed to be high, leading to relatively rapid transport of radionuclides released from the buffer through the geosphere to the biosphere. Due to the uncertainty in the properties of the fracture undergoing rock shear, the radionuclide concentration at the outer boundary of the buffer is conservatively set to zero. The geosphere transport resistance is set to WL/Q = 1,000, which is a factor of 50 lower than in the Base Case for an initial penetrating defect (case PD-BC). It is acknowledged that this reduction factor is somewhat arbitrary, and the geosphere transport resistance assumed in the case of canister failure due to rock shear will require better support in any future safety case.

A single canister failing due to rock shear at 70,000 years is postulated, this being the time of the next glacial retreat, assuming a repetition of the last glacial cycle. Thereafter, the supply of water to the canister interior is conservatively assumed to be unlimited and radionuclide release to the canister interior as well as solubility limitation of radionuclide concentrations are treated as in the Base Case for an initial penetrating defect. The ruptured canister is conservatively assumed to provide no resistance to radionuclide release. In reality, the ruptured canister may continue for some time to limit the ingress of water and the release of radionuclides, but, given the uncertainty in these factors, no credit is taken for these effects in the calculations.

The same steady groundwater flow and geochemical conditions are assumed in this Base Case as in the Base Cases for the other canister failure modes, although a variant case with dilute glacial meltwater assumed to be present at repository depth at all times subsequent to canister failure is also considered.

The consequences of multiple canister failures at around the same time can be obtained by multiplying the results for single canister failure by the number of canister failures postulated. As noted in Section 8.6, the expectation value of the number of canisters in the repository that could potentially be damaged by rock shear in the event of a large earthquake is currently estimated at 16 out of the total number of 3,000, although there are some significant uncertainties associated with these values that could lead to them giving either an underestimate or an overestimate of the actual likelihood of damage.

b. Calculational results

Figure 9-8 shows the releases from the geosphere to the biosphere as a function of time in the Base Case (RS-BC) expressed as WELL-2007 dose. Figure 9-9 shows time-dependent releases from the geosphere to the biosphere divided by the geo-bio flux constraints specified by the Finnish regulator and given in Table 2-1, and the sum of these releases over all radionuclides for which calculations were made.

There is a peak in dose of about 2×10^{-7} Sv for single canister failure that occurs shortly after canister failure at 70,000 years. The highest dose, however, occurs at later times, and is due to Ra-226. At a million years, an annual effective dose of a little over 10^{-6} Sv is calculated. The sum of time-dependent releases from the geosphere to the biosphere divided by their respective geo-bio flux constraints also has a peak shortly after 70,000 years (Figure 9-9). The highest values, however, again occur at later times and are due to Ra-226 (5×10^{-3} at a million years).

The pessimistic treatment of the geosphere means that there is only limited attenuation of near-field releases of many radionuclides by decay during geosphere transport in case RS-BC (see Section 7.2.4 of the Radionuclide Transport Report). This includes Ra-226 and its parent radionuclide Th-230, which are significantly attenuated by decay during geosphere transport in, for example, the Base Case for an initial penetrating defect (PD-BC). This explains the much higher release to the biosphere and WELL-2007 dose due to Ra-226 in case RS-BC compared with case PD-BC.

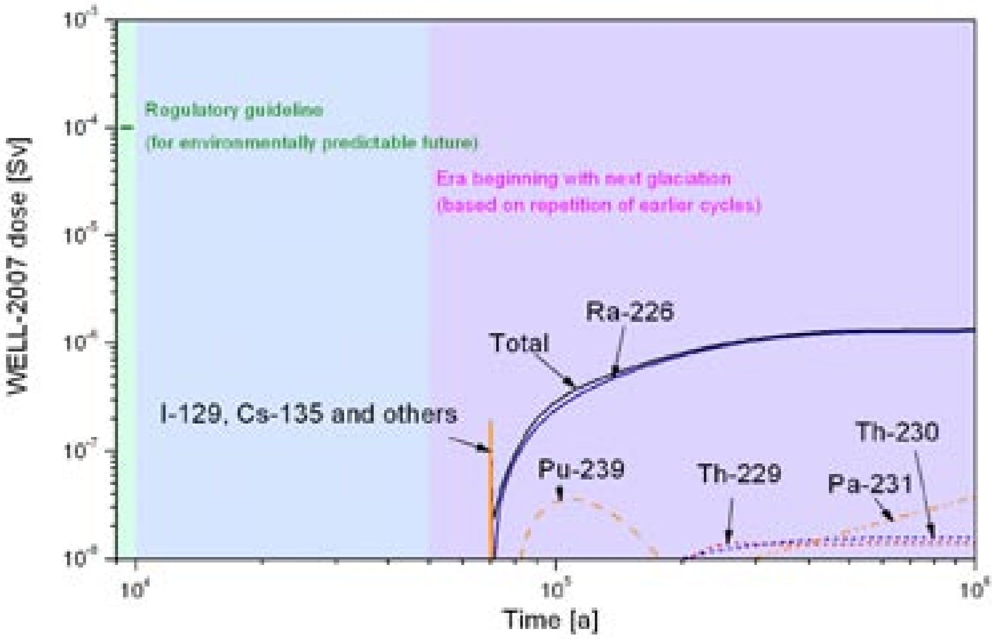

Figure 9-8. *WELL-2007 dose as a function of time in case RS-BC.*

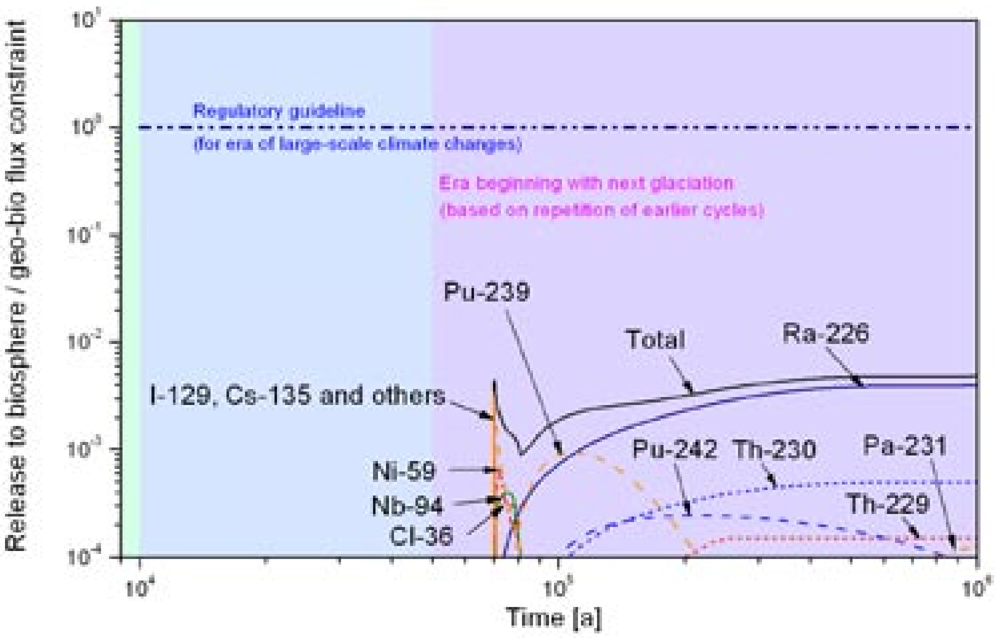

Figure 9-9. *Ratios of nuclide-specific activity releases to their respective geo-bio flux constraints in case RS-BC.*

9.7 Variant cases addressing issues related to the early evolution of a KBS-3H repository

The variant assessment cases addressing issues relevant to the early evolution of the repository with different significance to, or potential impact on, KBS-3H compared with KBS-3V are:

- PD-HIDIFF, which illustrates the impact of increased diffusion in the buffer (e.g. due to piping and erosion during early evolution).

- PD-SPALL, PD-FEBENT1, PD-FEBENT2, and PD-FEBENT3, which illustrate the impact of various processes that could significantly affect the transport properties of the buffer/rock interface.
- PD-EXPELL, which illustrates the impact of the expulsion of water and radionuclides from the interior of a failed canister by gas.

Given their particular significance to the present safety assessment, with its focus on differences between KBS-3V and KBS-3H, these assessment cases are described in some detail in the following sections. For a still more detailed description of all assessment cases, the reader is referred to the Radionuclide Transport Report.

9.7.1 Increased diffusion in the buffer (e.g. due to piping and erosion)

An increase in buffer diffusion coefficient due, for example, to piping and erosion and buffer density loss could perturb radionuclide transport across the buffer in the event of canister failure. Case PD-HIDIFF is an assessment cases in which an increased buffer diffusion coefficient is assumed in modelling radionuclide transport through the buffer in the event of radionuclide release from a canister with an initial penetrating defect.

In this assessment case, the effective diffusion coefficient for the various radionuclide species in the buffer is increased from their Base Case values to a species-independent, value of 8.6×10^{-10} m^2 s^{-1}, which is considered to be a bounding value at the high end of the range of possibilities (see Section 5.6.3 of the Radionuclide Transport Report[25]).

In Figure 9-10, releases from the geosphere to the biosphere, divided by the geo-bio flux constraints specified by the Finnish regulator, are compared for the Base Case (PD-BC) and for case PD-HIDIFF. Total release and the release of C-14, I-129 and Cs-135 are shown. These are the radionuclides that dominate total release at various times in both cases.

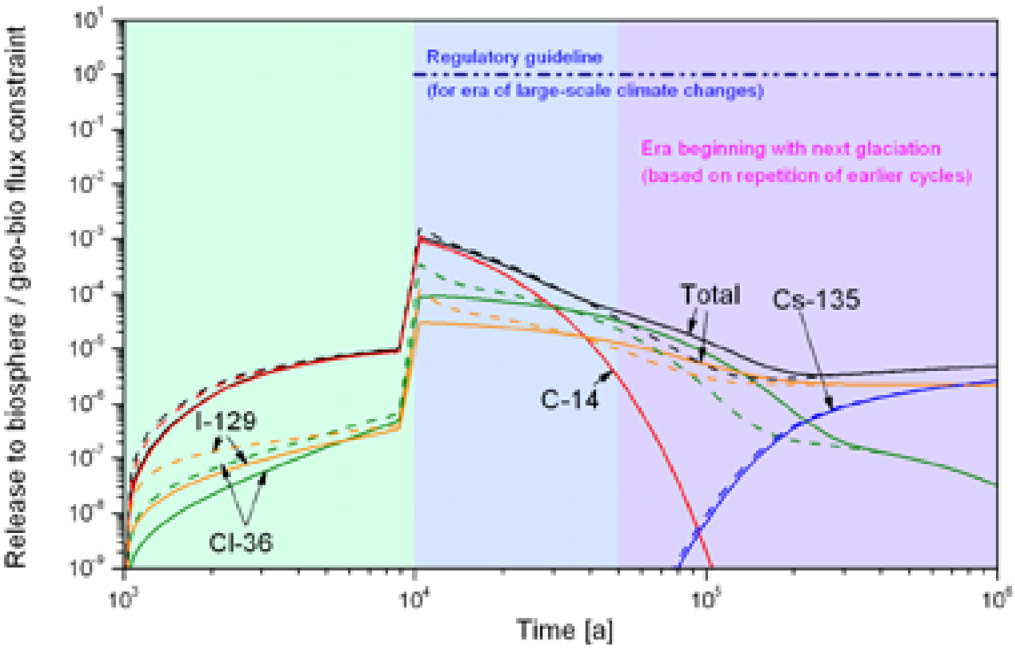

Figure 9-10. *Ratios of nuclide-specific activity releases to their respective geo-bio flux constraints for selected radionuclides in cases PD-HIDIFF (solid lines) and the Base Case PD-BC (dashed lines).*

[25] Major ions found in groundwater have diffusion coefficients in free water in the range 10^{-9} to 2×10^{-9} m^2 s^{-1} at 25°C, see, e.g. Section 3.4 in /Freeze and Cherry 1979/. The transport porosity for this case is taken to be 0.43 for all migrating species. The effective diffusion coefficient is calculated from the diffusion coefficient in free water (2×10^{-9} m^2 s^{-1}) multiplied by the transport porosity.

There are two competing effects, such that releases of some radionuclides at certain times are decreased in case PD-HIDIFF compared with the Base Case, whereas other releases are virtually unchanged or may be increased. The increased diffusion coefficient in the buffer in case PD-HIDIFF means that radionuclides migrate further in the axial direction along the drift, as well as radially towards the drift wall. This provides more dilution of radionuclides in buffer pore water, which tends to decrease peak releases and extend the tailing parts of the release curves, especially in the case of non-solubility-limited anions, such as C-14, Cl-36 and I-129, since, unlike the Base Case (PD-BC) and the majority of other cases, anionic exclusion is not considered in the buffer in case PD-HIDIFF, consistent with the assumption of a bounding value for the diffusion coefficient at the high end of the range of possibilities. On the other hand, the increased diffusion coefficient in the buffer in case PD-HIDIFF also reduces the minimum transport time across the buffer. This leads to less radionuclide decay during transport, and tends to increase near-field release rates.

Overall, the effect of increased diffusion in the buffer is small, and the peak release summed over all radionuclides is virtually unchanged with respect to the Base Case (PD-BC).

9.7.2 Perturbations to the buffer/rock interface

There are a number of processes identified in Section 7.1 that could significantly affect the transport properties of the buffer/rock interface. These are:

- thermally-induced rock spalling,
- the presence of potentially porous or fractured corrosion products in contact with the drift wall in relatively tight drift sections,
- chemical interaction of the buffer with these corrosion products,
- chemical interaction of the buffer with high-pH leachates from cementitious components.

In the Base Cases for the different canister failure modes, it is assumed that the impact of these processes on radionuclide release and transport is negligible. The degree to which system properties will, in reality, be affected and the spatial extent of the effects is, however, highly uncertain. Thus, four variant cases are considered in which the impact of these processes is assumed to be more significant. All four cases are similar, in that the perturbing processes are assumed to create a high-permeability zone at the interface. The extent of the zone, and the groundwater flow within the zone, is, however, case dependent.

Case PD-SPALL addresses thermally induced rock spalling in a relatively tight drift section, where buffer swelling pressure at the drift wall is not developed rapidly enough to prevent this process from taking place. It could also be considered to address the impact of porous iron corrosion products being in direct contact with the drift wall – this is also a possibility in relatively tight drift sections (Section 5.5.3 in the Evolution Report). Although the hydraulic conductivity of the host rock near the drift wall in affected drift sections is assumed to be substantially increased by thermally induced spalling, the pressure on the drift wall exerted by the distance blocks is assumed to suppress spalling and will prevent the formation of continuous flow and transport pathways along the drift (Section 7.1.6).

Cases PD-FEBENT1, PD-FEBENT2 and PD-FEBENT3 address chemical interaction of the buffer with the iron of the supercontainer (or with high-pH leachates from cementitious repository components). In case PD-FEBENT1, the extent of the affected buffer zone is assumed to be limited to a region around the supercontainer of vanishingly small thickness but with very high (effectively infinite) hydraulic conductivity. In case PD-FEBENT2, the affected buffer zone is assumed to extend across 10% of the entire thickness of the buffer (4 cm). This is consistent, for example, with the results of reactive transport modelling, which indicate that the extent of the zone potentially undergoing mineral transformation due to iron/bentonite interaction is likely to remain spatially limited (a few centimetres) for hundreds of thousands of years or more /Wersin et al. 2007/. In case PD-FEBENT3, the affected buffer zone is conservatively assumed to extend across half the entire thickness of the buffer (20 cm).

In Figure 9-11, releases from the geosphere to the biosphere, divided by the geo-bio flux constraints specified by the Finnish regulator and then summed over all radionuclides, for all four cases addressing perturbations to the buffer/rock interface are compared with the Base Case for a canister with an initial penetrating defect (PD-BC).

Releases in case PD-SPALL are slightly reduced with respect to the Base Case, since, although the buffer/rock interface is unfavourably perturbed in case PD-SPALL, thermally-induced rock spalling is assumed to affect only a relatively tight drift section. The release maxima in cases PD-FEBENT1, PD-FEBENT2 and PD-FEBENT3 are similar to each other (the thickness of the perturbed interface zone is an insensitive parameter), and increased by about an order of magnitude with respect to the Base Case.

9.7.3 Gas expulsion

In the Base Case for the different canister failure modes, it is assumed that gas generated inside a defective canister, principally hydrogen from the corrosion of steel, has no impact on radionuclide transport. As discussed in Section 7.1.7, however, if there is an initial penetrating defect and the defect is unfavourably located on the underside of the canister, the possibility of gas-induced release of contaminated water from the canister interior to the buffer cannot be excluded. This situation is addressed in case PD-EXPELL, where it is assumed that a gas-driven water pulse, beginning at 2,800 years after deposition and lasting for a further 1,300 years, propels water from the canister interior through the buffer to the fracture. These assumptions are based on the most pessimistic case from a range of model calculations of the fate of water/vapour/gas and radionuclides described in Section 2.5 of the KBS-3H Process Report.

Figure 9-12 gives the calculated time-dependent WELL-2007 dose for this case (rather than releases from the geosphere to the biosphere divided by the geo-bio flux constraints, since the maximum occurs during the environmentally predictable future; all results are, however, given in the Radionuclide Transport Report). The dose maximum, which occurs shortly after the start of water expulsion, takes a value of 2×10^{-7} Sv and is therefore significantly increased relative to the Base Case (PD-BC).

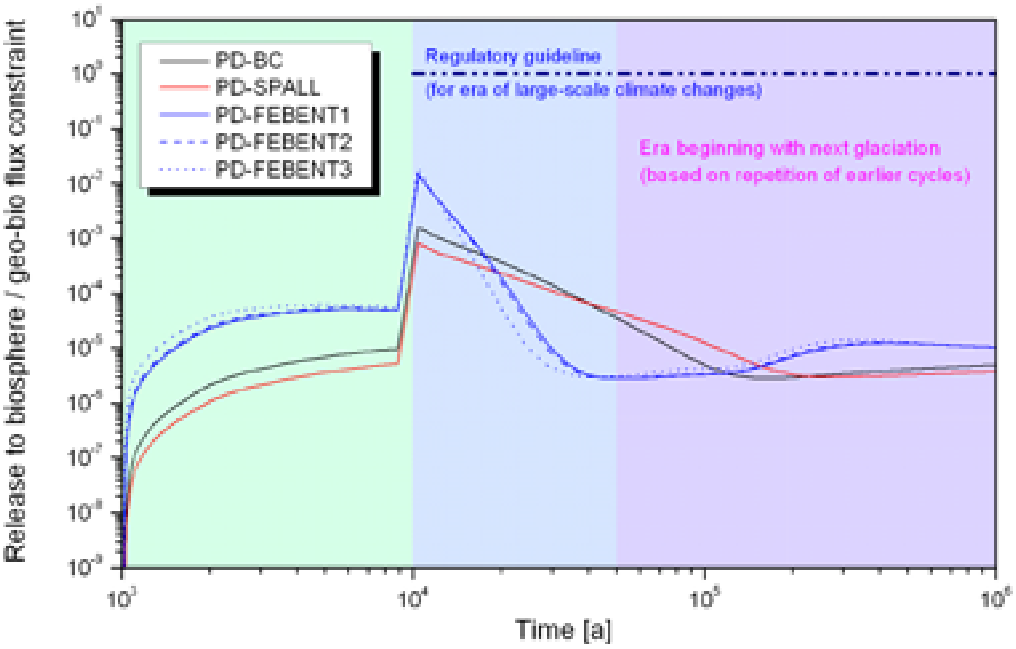

Figure 9-11. *Releases from the geosphere to the biosphere, divided by the geo-bio flux constraints specified by the Finnish regulator and then summed over all radionuclides, in the Base Case (PD-BC) and in all four cases addressing perturbations to the buffer/rock interface.*

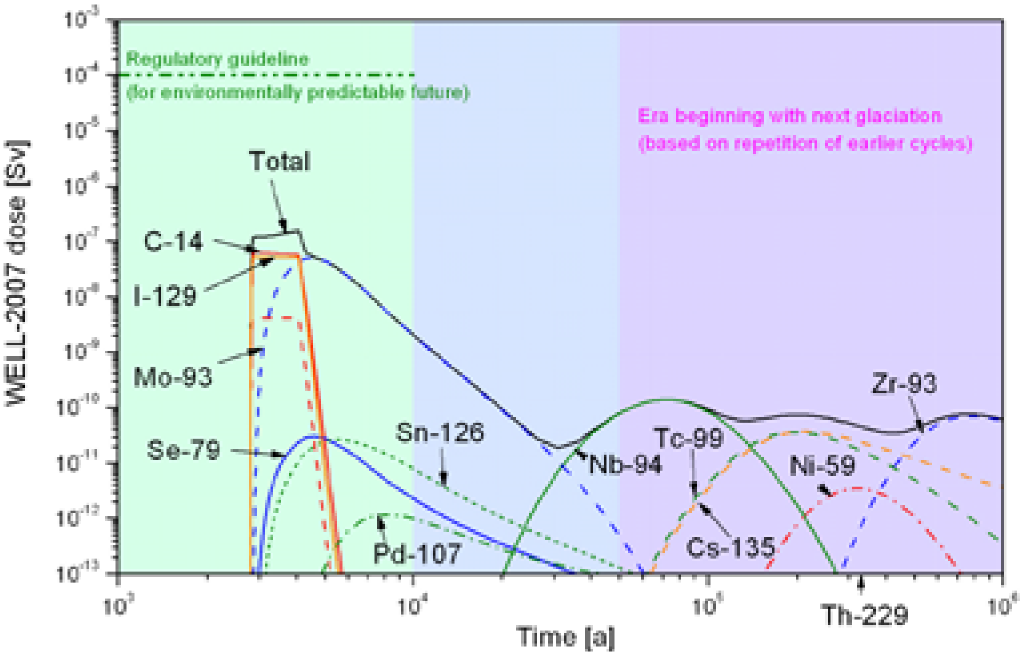

Figure 9-12. WELL-2007 dose as a function of time in case PD-EXPELL.

The likelihood of this case and the possibility gas expulsion could occur from multiple canisters at similar times have not been evaluated. It is, however, emphasised that an actual defect may be located anywhere around the canister and no significance should be attached to the depiction of the location of the defect at the upper side of the canister in Figure 9-1. Furthermore, as in all cases addressing an initial penetrating defect, the water inflows may be much reduced due to sealing of the hole by bentonite and corrosion products, making this variant case less likely.

9.8 Evaluations of annual landscape dose

There is a calculated release to the biosphere within the time frame from emplacement up to several thousand years in the future only in the assessment cases assuming an initial penetrating defect (and excluding cases PD-HIDELAY and PD-VOL-2). It is in these cases that dose assessments are explicitly required by regulations, and annual landscape doses have therefore been estimated for the environmentally predictable future (the period up to 10,000 years in the future). Results from six representative assessment cases are included here. Other cases assuming an initial penetrating defect have been treated by scaling approaches or qualitative arguments, as described in the Biosphere Analysis Report /Broed et al. 2007/. Other canister failure modes occur after the "environmentally predictable future" and so no evaluation of annual landscape dose is required.

Figure 9-13 shows the calculated annual landscape dose to the most exposed individual in the six assessment cases.

There is a sharp increase in annual landscape dose starting at about 9,000 years in cases PD-BC, PD-FEBENT3, PD-HISAL, PD-LOGEOR, associated with the loss of transport resistance of the defect (which occurs earlier in case PD-LODELAY). As noted earlier, this loss of transport resistance is, in fact, assumed to occur at 10,000 years in the near-field model of these cases. This is when the calculation of annual landscape dose is terminated. The earlier calculated onset of the dose increase is an artefact of the finite time step size in geosphere transport modelling.

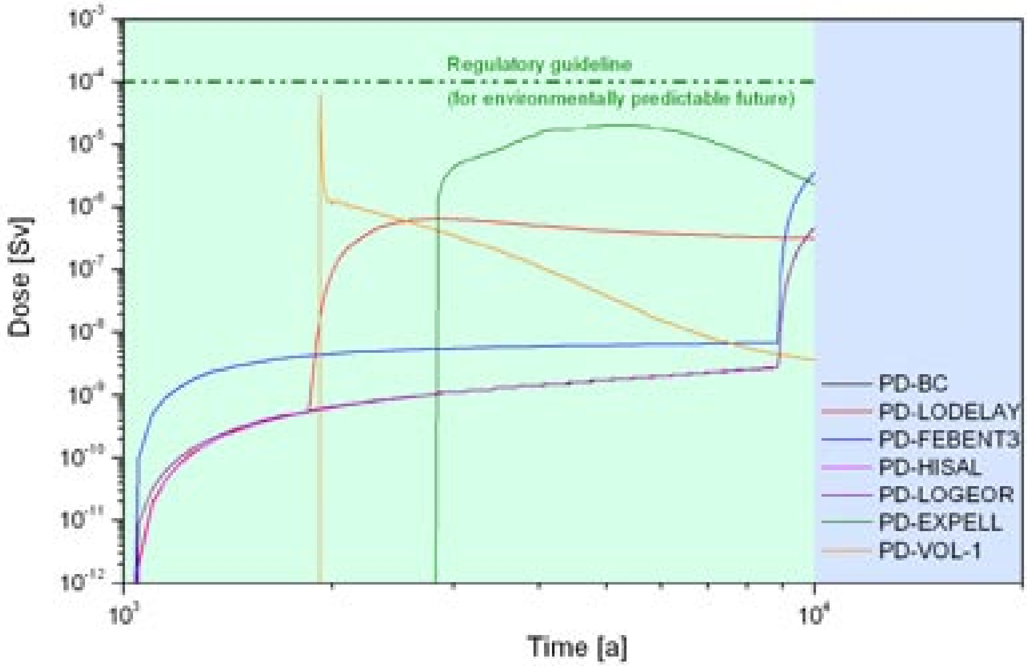

Figure 9-13. *Annual landscape dose to the most exposed individual due to potential releases from the repository in seven most representative assessment cases for a canister with an initial penetrating defect (results for PD-BC, PD-HISAL and PD-LOGEOR approximately coincide).*

Of the seven cases calculated, the highest calculated annual landscape dose occurs in case PD-VOL-1, with a maximum of about 6×10^{-5} Sv occurring at about 1,900 years. It should, however, again be noted that only a single failed canister is considered in each case. The issue of multiple canister failures is discussed briefly in Section 9.9.4.

9.9 Overview of assessment case results and evaluation of compliance

9.9.1 Annual landscape dose maxima

The highest calculated annual landscape dose for the most exposed individual is about 6×10^{-5} Sv, and arises in the assessment case PD-VOL-1. This is a little less than a factor of two below the regulatory constraint of 10^{-4} Sv. For all other assessment cases, the maxima range from 5×10^{-7} to 2×10^{-5} Sv, and are thus around an order of magnitude or more below the regulatory constraint.

9.9.2 WELL-2007 dose maxima

Figure 9-14 shows the magnitudes of the calculated WELL-2007 dose maxima in all the assessment cases considered in the safety assessment. The cases are arranged in two groups: at the top are those cases for which the dose maximum occurs in the era of extreme climate changes, and at the bottom (the last four) are those cases in which the dose maximum occurs in the environmentally predictable future, which is taken to extend to 10,000 years post closure. Within each group, cases are arranged in order of descending magnitude of the dose maximum. The regulatory guideline of 10^{-4} Sv is shown across the entire million year assessment period, although it is applicable only during the environmentally predictable future, and so is plotted as a dashed line, rather than as a solid line, at later times.

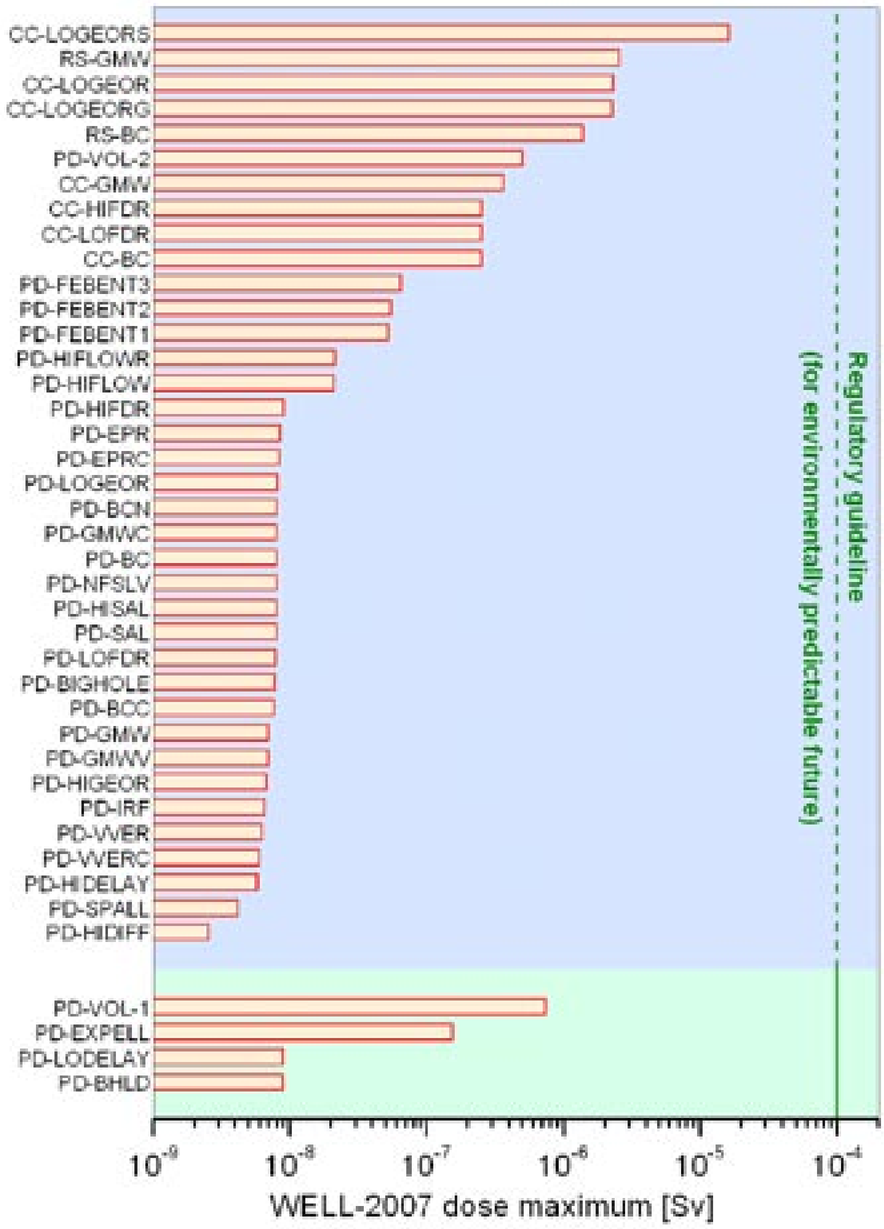

Figure 9-14. *Calculated dose maxima in all assessment cases. Green background shading indicates the maxima that occur within the first 10,000 years post closure, which is interpreted in the present study as the "environmentally predictable future".*

The figure shows that, in all cases, the calculated dose maxima are below the regulatory guideline of 10^{-4} Sv per year (although, as noted above, in most cases the maxima occur beyond the time frame in which this guideline applies). The highest calculated dose maxima are in the cases involving canister failure by corrosion or by rock shear, in which there is no assumed subsequent period in which the failed canister provides a transport resistance and the geosphere transport resistance is assumed to be lower than in other cases, allowing sorbing radionuclides, such as Th-230 and Ra-226, to migrate across the geosphere without substantial attenuation by radioactive decay. The entire instant release fraction from the fuel is thus released to the buffer as a pulse in these cases, whereas there is more spreading in time in cases involving a small, initial penetrating defect.

In cases PD-EXPELL, PD-VOL-1, PD-LODELAY and PD-BHLD, in which calculated dose maxima occur in the environmentally predictable future when the regulatory dose guideline applies, the maxima are 2–4 orders of magnitude below the guideline.

The differences between the WELL-2007 doses and the landscape doses are most likely due to different conceptual assumptions, particularly those concerning the biosphere, e.g. accumulation in the biosphere, identification of the most exposed subgroup of the population, radionuclide-specific biosphere transport processes. Work is still ongoing to analyse the different results and identify key assumptions affecting the doses at different end points.

In many cases assuming a canister with an initial penetrating defect, the release maxima occur shortly after 10,000 years – i.e. in the era of large-scale climate changes – 10,000 years being the time assumed for loss of transport resistance of the defect in most cases. It is important to note, however, that uncertainty in the time when an initial penetrating defect loses its transport resistance is such that the maxima could equally well occur earlier – i.e. in the environmentally predictable future – or much later, up to a million years in the future.

9.9.3 Geo-bio flux maxima

Figure 9-15 shows the magnitudes of the summed release maxima in all the assessment cases considered in the safety assessment. The cases are again arranged in two groups: at the top are those cases where the release maximum occurs in the era of extreme climate changes, and at the bottom (the last four) are those cases where the release maximum occurs in the environmentally predictable future. Within each group, cases are arranged in order of descending magnitude of the release maximum.

The highest calculated summed release maximum occurs in case CC-LOGEORS, i.e. canister failure by copper corrosion, in association, for example, with an influx of glacial meltwater and loss of buffer mass by chemical erosion, coupled to an assumption of low transport resistance and saline geochemical conditions in the geosphere at later times. In this case, the summed release maximum is more than an order of magnitude below the regulatory guideline. Nevertheless, there are significant uncertainties associated with this scenario, i.e. whether substantial buffer mass loss by chemical erosion could occur at all, and, if it does, the number of canister positions that are likely to be affected (case CC-LOGEORS deals with only a single canister failure). The development of a better understanding of chemical erosion is a priority for future work for both the KBS-3H and KBS-3V repositories.

The next highest calculated release maxima occur in cases PD-VOL-1 and PD-VOL-2, i.e. expulsion of C-14 in volatile form by repository-generated gas through an initial penetrating defect (see Section 5.9 of the Radionuclide Transport Report). Results in these cases have been averaged over a 1,000 year interval, as allowed by Finnish regulations. Without such averaging, radionuclide release to the biosphere is close to the regulatory guideline, as illustrated in Figure 9-16.

9.9.4 Compliance with regulatory criteria in different time frames and the issue of multiple canister failures

Overall, the release and transport calculations indicate compliance with Finnish regulations both for the "environmentally predictable future" and for later times, assuming single canister failure by any of the identified modes.

The probability of multiple canisters failures at similar times has been evaluated only in the case of canister failure due to rock shear – the expectation value of the number of canisters in the repository that could potentially be damaged by rock shear in the event of a large earthquake has been estimated to be 16 out of the total number of 3,000 canisters (Section 8.6). The geo-bio flux constraint will still be met in cases RS-BC and RS-GMW if 16 canisters fail in the event of a single large earthquake. However, the number of canisters that might fail over a million year time frame (and contribute to the Ra-226 dose at a million years) has not so far been evaluated (see, however, the discussion of the hazards arising from radionuclide releases in Chapter 10, where a bounding analysis with all canisters failing is presented).

More generally, a key issue for future safety assessments will be to better quantify the probability of the failure of several canisters at similar times.

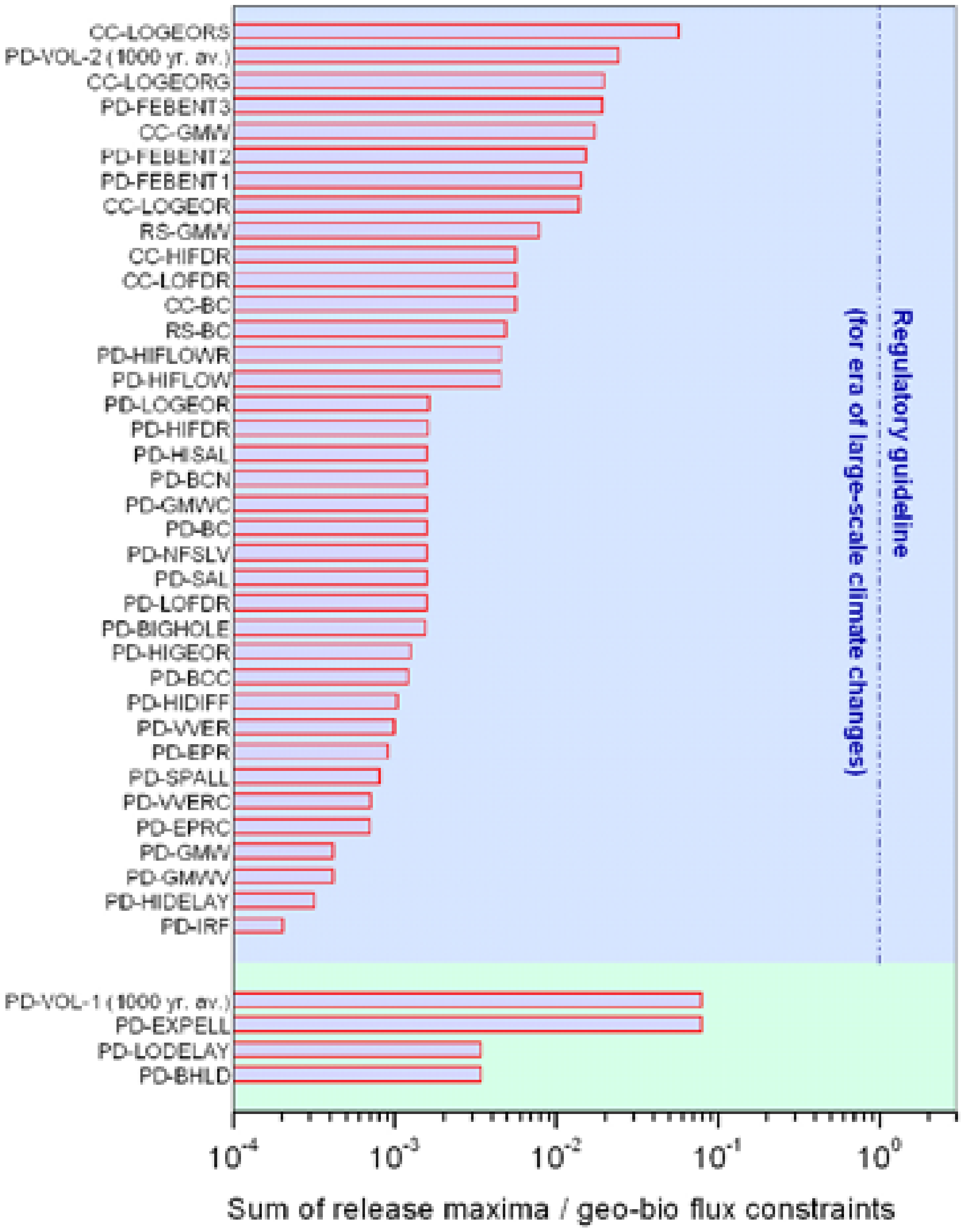

Figure 9-15. *Calculated summed release rate maxima in all assessment cases. Before carrying out the summation, nuclide-specific activity releases are divided by their respective geo-bio flux constraints. Green background shading indicates the maxima that occur within the first 10,000 years post closure, which is interpreted in the present study as the "environmentally predictable future". Results in cases PD-VOL-1 and PD-VOL-2 are averaged over a 1,000 year interval (see text).*

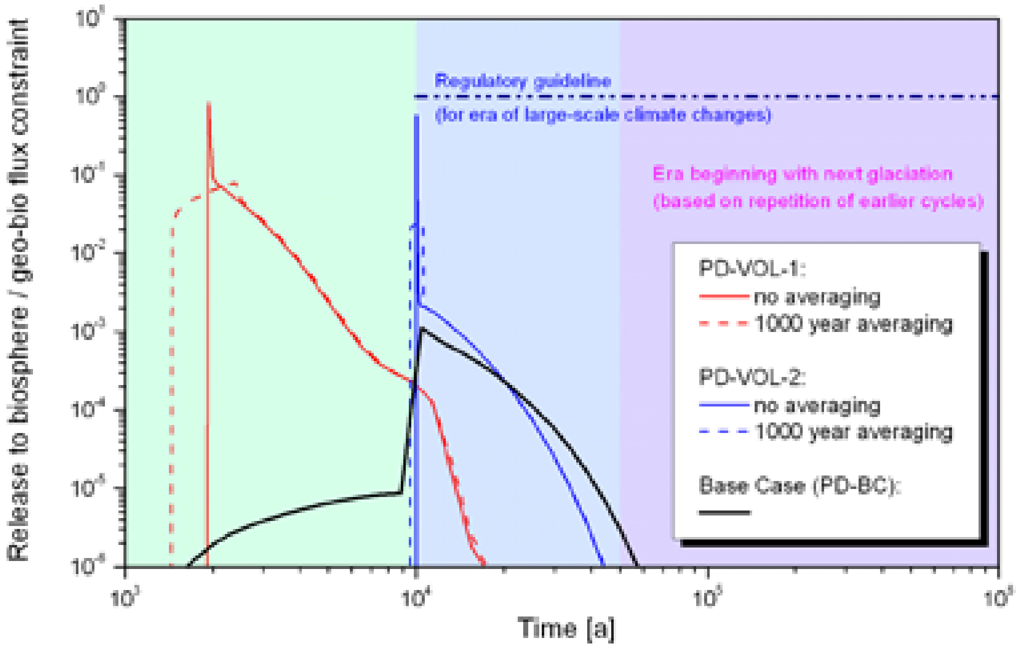

Figure 9-16. *Ratios of C-14 activity release to the geo-bio flux constraint in the Base Case (PD-BC) and in cases PD-VOL-1 and PD-VOL-2.*

9.10 Comparison of results with TILA-99, SR-Can and other safety assessments

A comparison of the results of the KBS-3H safety assessment calculations with those of TILA-99 and SR-Can given in the Complementary Evaluations of Safety Report confirms that they are consistently in the same range and well below the regulatory dose criterion. Moreover, where there are differences between results, it is possible to identify the reasons. For example, SR-Can results have noticeable releases of Ra-226 to the biosphere which support an increase in dose after about 100,000 years in several calculational cases. There is no similar contribution from Ra-226 seen in KBS-3H assessment results in most cases because these releases in SR-Can reflect the influence of a small number of fast geosphere pathways that arise in the stochastic geosphere model adopted. A similar contribution from Ra-226 only occurs in KBS-3H in the cases where the single representative geosphere flow path is assigned parameter values which make its properties more similar to the SR-Can fast pathways as, for example, in the rock shear case (Figure 9-9).

10 Complementary evidence and arguments for safety

10.1 Evidence from natural and anthropogenic analogues

Evidence and arguments to support the long-term performance of the repository, and to better understand the various materials present in the repository system, have been gained from studies of natural and anthropogenic analogues, as discussed in Section 3.3 of the Complementary Evaluations of Safety Report (and references therein). These analogues have included uranium ore deposits, native copper occurrences, copper and iron archaeological finds and deposits of bentonite and related clay materials.

Natural uranium deposits, such as those at Cigar Lake in Canada /Cramer and Smellie 1994/, Palmottu in Finland /Blomqvist et al. 2000/, Oklo in Gabon /e.g. Brookins 1990/, Poços de Caldas in Brazil /Chapman et al. 1992/ and Koongarra in Australia /Duerden 1990/, have been used to justify the assumption of a low rate of dissolution of spent fuel in safety assessment by comparing the UO_2 of the spent fuel to the naturally-occurring uraninite. These and other uranium deposits have also been useful to study other long-term processes such as:

- Redox processes and their role in radionuclide mobilisation and retardation.
- Radionuclide speciation and solubility, including the formation and behaviour of colloids, in a wide variety of groundwater conditions.
- Retardation processes affecting mobilised radionuclides, including sorption on fracture minerals and diffusion in the matrix, in a range of different rock types.
- Radionuclide mobility facilitated by colloids and microbial populations.

The Cigar Lake uranium deposit /Cramer and Smellie 1994/ is particularly interesting as it provides a large-scale analogue for a generic geological repository in hard, fractured formations. It is particularly relevant to spent nuclear fuel repositories because of the large amount of uranium, and very high uranium concentrations present, up to around 55% /Cramer and Smellie 1994/[26]. It is also located at a depth similar to that of the Olkiluoto repository (–450 m). The uranium ore body, surrounded by a clay-rich halo, is located on the unconformity between old (Proterozoic) sandstones and underlying crystalline basement rocks, in an environment with active groundwater circulation.

Although the Cigar Lake uranium ore body was formed about 1,300 million years ago, there are no geochemical signatures at the ground surface, illustrating the isolation capacity of the clay-rich halo and the sedimentary host rock above the ore body.

The long-term durability of native copper in relevant conditions is illustrative evidence for the long-term stability of copper canisters. Several examples of copper occurrences have corroborated the expectation that the oxygen-free and reducing conditions in the repository will preserve the copper canisters. Indeed, native copper has persisted for millions of years in several geological environments /e.g. Marcos 1989/ such as:

- In sedimentary rocks: Keweenaw Peninsula, Lake Superior region, Michigan, U.S.; Corocoro, Bolivia; south Devon, United Kingdom.
- In basaltic lavas: Keweenaw Peninsula; Appalachian States from central Virginia to southern Pennsylvania, U.S.; Coppermine River area, NWT, Canada; Dalane, Norway.

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[26] Direct comparison of a uranium deposit with spent fuel is however not straightforward. The Cigar Lake ore deposit comprises uraninite ($UO_{2.2}$) which is crystallographically the same as crystalline UO_2, but includes U(VI) atoms in its lattice. The presence of U(VI) causes the solubility of uraninite to be somewhat higher than that of UO_2 in spent fuel.

- In granitic rocks: Hyrkkölä and Askola, Finland.
- In the oxidised zones of sulphide deposits (many places in the world, including Finland; the deposits in Chile may be the best known) and in swamps.

Complementing studies on these and other native copper occurrences in oxygen-free conditions, an archaeological analogue has provided useful support for the corrosion resistance and longevity of copper in oxidising conditions. The Kronan bronze cannon /Hallberg et al. 1987/ from the Swedish man-o-war "Kronan" was partially buried muzzle down in shallow marine clay when the ship sank in 1676. The Kronan cannon has a very high copper content (96.5%, with 3.3% tin and 0.5% iron). The head of the cannon protruded from the clay and was in direct contact with seawater. The average corrosion rate (0.15 microns per year over 300 years) of the buried part of the cannon was determined based on the time since the warship sank. Thus, the corrosion rate has remained low even in oxidising conditions that would be expected to be favourable for copper corrosion. The oxidation products identified (CuO_2, Fe_3O_4) are in agreement with models for corrosion in moderately oxygenated water – rather different to the expected repository environment but a useful indicator for the possible case of oxygenated groundwaters reaching repository depth due to glaciation[27].

Natural systems have another important role to play with respect to the long timescales involved in radioactive waste disposal. The Finnish regulatory dose criteria /STUK 2001/ acknowledge that, due to increasing uncertainties about the surface environment in the future, the calculation of dose to a human population becomes less meaningful in the long term, especially after the onset of a future glaciation. Other calculated performance and safety indicators can, however, be used to complement calculated doses. Such indicators can show, for example, how the potential toxicity of spent fuel due to ionising radiation compares with that of naturally occurring radioactive materials, such as uranium ore bodies (Section 10.3), and how calculated radionuclide releases from the repository compare with naturally occurring radionuclide fluxes (Section 10.4).

With some natural and archaeological analogues, it is not possible to extract detailed technical information but, even so, the support these analogues can convey should not be underestimated: that the Cigar Lake uranium deposit has no surface geochemical anomaly to indicate its presence after 1.3 billion years of existence is evidence of the most direct kind that a system with a suitable hydrogeological and geochemical environment can provide safety over the timescales required.

10.2 Evidence for site suitability

The geological stability of the site is a pre-requisite to ensure that the host rock performs the safety functions described in Section 4.3. The location of a site in the Fennoscandian shield, and especially in Finland, is advantageous with respect to geological stability, in terms of seismic (and volcanic) activity and in terms of tectonic uplift /Marcos et al. 2007/. This is because it is remote from active plate margins and on-going mountain building.

The Olkiluoto site is located on the 1,800–1,900 million years old shield area of southern Finland. Like Finland in general, it has low seismic activity. Earthquakes in Northern Europe since 1375 are shown in Figure 10-1. The figure shows that the density and magnitude of earthquakes in Finland is lower than in many other areas. Earthquakes magnitudes in Finland have never reached 5 on the Richter scale since records began in the 1880s /Marcos et al. 2007/ and references therein). At a regional level, seismic activity in the Olkiluoto region is currently low (see e.g. /La Pointe and Hermanson 2002, Enescu et al. 2003, Saari 2006/). Seismic studies at Olkiluoto show negligible rock movements /Andersson et al. 2007/. However, greater seismic activity in the future cannot be excluded, for example, as part of future post-glacial readjustment.

[27] The future intrusion of oxygen-rich glacial meltwater into the deep groundwater system at Olkiluoto is unlikely (Section 7.2.4), but it is also identified as an issue where additional information is needed (Section 11.2.1).

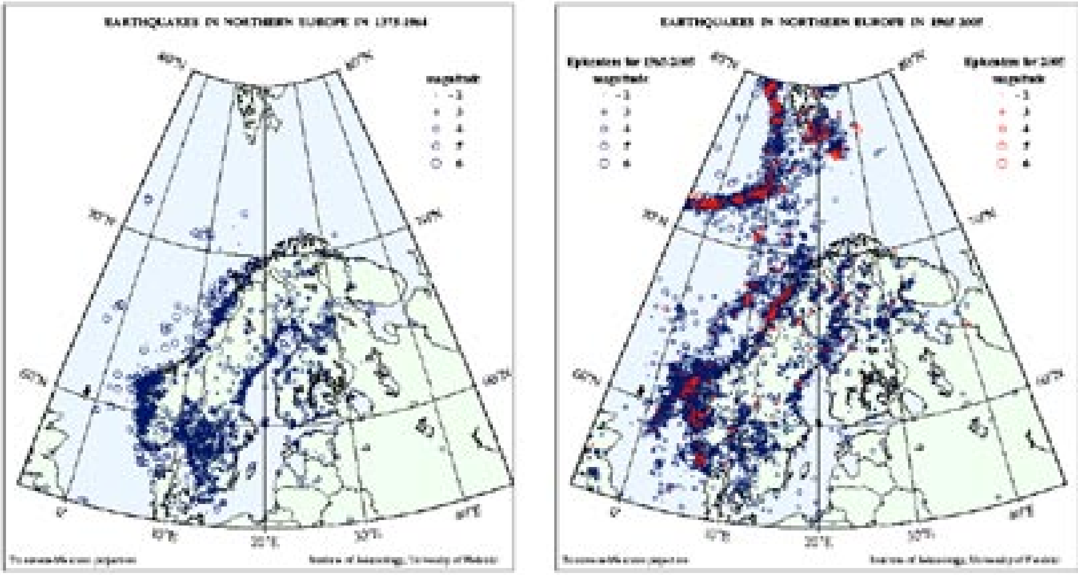

Figure 10-1. Earthquakes in Northern Europe for 1375–1964 (left) and for 1965–2005 (right). Note that the density and magnitude of earthquakes in Finland is lower than in other sites in Northern Europe. (Source: University of Helsinki[28]).

The Olkiluoto site is not currently affected by tectonic uplift[29], and there is evidence of regional stability over millions of years, with no suggestion that this situation will be disrupted by changes in plate tectonics in the next few million years.

10.3 Radiotoxicity of the spent fuel

Radiotoxicity is a measure of the potential toxicity due to ionising radiation (often referred to as the "hypothetical dose") following ingestion (or inhalation) of a radionuclide. The radiotoxicity of a radioactive material, $RT(t)$ [Sv], is here defined as the hypothetical dose at time t resulting from ingestion of the activity $A_j\,(t)$ [Bq] of radionuclide j /Nagra 2002, Hedin 1997/. Thus:

$$RT(t) = \sum_j A_j D_j \qquad \text{(Equation 10-1)}$$

where D_j [Sv/Bq] is its dose coefficient for ingestion.

Further, the radiological hazard is sometimes expressed in terms of a "radiotoxicity index" or $RTI(t)$ [–] /Nagra 2002, Hedin 1997/, here defined as the hypothetical dose at time t resulting from ingestion of the activity $A_j\,(t)$ [Bq] of radionuclide j, divided by 10^{-4} Sv (derived from the Finnish regulatory dose limit for the first several thousand years):

$$RTI(t) = \frac{\sum_j A_j(t) D_j}{10^{-4}\,Sv} \qquad \text{(Equation 10-2)}$$

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[28] Source for left figure: http://www.seismo.Helsinki.fi/bulletin/list/catalog/histomap.html. Source for right figure: http://www.seismo.Helsinki.fi/bulletin/list/catalog/instrumap.html.

[29] As mentioned in Section 4.3, tectonic uplift is not the same as post-glacial (isostatic) uplift, which is high in this region.

Consideration of radiotoxicity or radiotoxicity index of the spent fuel compared with naturally occurring radioactive materials, such as uranium ore bodies, allows some interesting points to be demonstrated. In particular, the steep decline in the radiotoxicity of the waste over the first thousands of years due to decay of fission products and activation products contrasts strongly with naturally occurring U and Th. These natural elements have primordial isotopic compositions (i.e. unaffected by man-made processes) and very long half-lives. Thus, over the million year period of the assessment, they show no significant change in radiotoxicity. By about 100,000 years after repository closure, the activity of the spent fuel has decayed such that it has about the same radiotoxicity as the larger quantity of uranium which was used to produce the original fuel. This comparison is not made in order to suggest that the spent fuel is "safe" at this time, but rather to illustrate that the magnitude of the hazard (at least the potential hazard due to ingestion or inhalation) has reduced to a level comparable with that of a naturally occurring uranium ore body of a similar extent.

10.4 Hazards arising from repository radionuclide releases

Calculated doses due to the repository can be compared not only with regulatory guidelines but also with doses arising from natural background radiation or consumption of normal foodstuffs. The calculated doses arising from the repository releases are shown to be insignificant by comparison with the doses incurred from consumption of ordinary drinking water: the calculated maximum dose in the base case for a canister with a penetrating defect (assessment case PD-BC, see Figure 9-13) is more than 3 orders of magnitude smaller than the dose arising from drinking the same amount (500 litres per annum) of tap water in the municipality of Eurajoki, where the Olkiluoto site is located (Table 6-1, /Neall et al. 2007/). Even in the pessimistic rock shear failure base case (RS-BC, see Figure 9-15), in which the geosphere barrier is largely circumvented, the maximum release dose is about 50 times smaller than that due to drinking 500 litres of Eurajoki tap water (Table 6-1, /Neall et al. 2007/).

The issue of multiple canister failures at similar times has not been addressed in the present safety assessment, except in the case of canister failure due to rock shear in the event of a single large earthquake. A "worst case scenario" can, however, be assessed in a bounding case in which, contrary to expectations, all canisters fail within the assessment period, and the failure mode circumvents most of the barrier capacity of the geosphere (as in the case of failure due to rock shear). Even in this extreme, hypothetical case, the maximum annual effective dose arising from all 3,000 canisters is around 4 mSv per year at a million years[30]. This is about the same as the annual exposure of the Finnish population to natural and anthropogenic radiation sources (including all exposures). Natural background radiation in Finland is dominated by indoor inhalation of radon. The average annual concentration of radon gas in dwellings is about 120 Bq/m^3 (www.stuk.fi), which probably is the highest in the world, and gives rise to an average annual effective dose of about 2mSv.

In addition to assessment endpoints based on radiation doses (see Section 9.5), further evaluations of the repository releases are described in the Complementary Evaluations of Safety Report on the basis of radiotoxicity flux. The radiotoxicity flux is a measure of the hazard created by a flux of radionuclides across a defined interface within a given period. For these evaluations, a hypothetical interface of 1 km^2 above the repository was assumed. The total calculated annual flux from the repository through the geosphere to the biosphere was assumed to pass across this interface. This simple construction allows the repository releases to be compared with natural radionuclides moving in groundwater across this hypothetical interface at repository depth or with radionuclides released by weathering of the bedrock over the same 1 km^2 but at the surface. These comparisons are shown in Figure 10-2 along with the radiotoxicity fluxes calculated for surface discharge of groundwater at Olkiluoto in two distinct areas ('north' and 'south').

[30] This is in agreement with the similar case calculated for the earlier TILA-99 safety assessment /Vieno and Nordman 1999/.

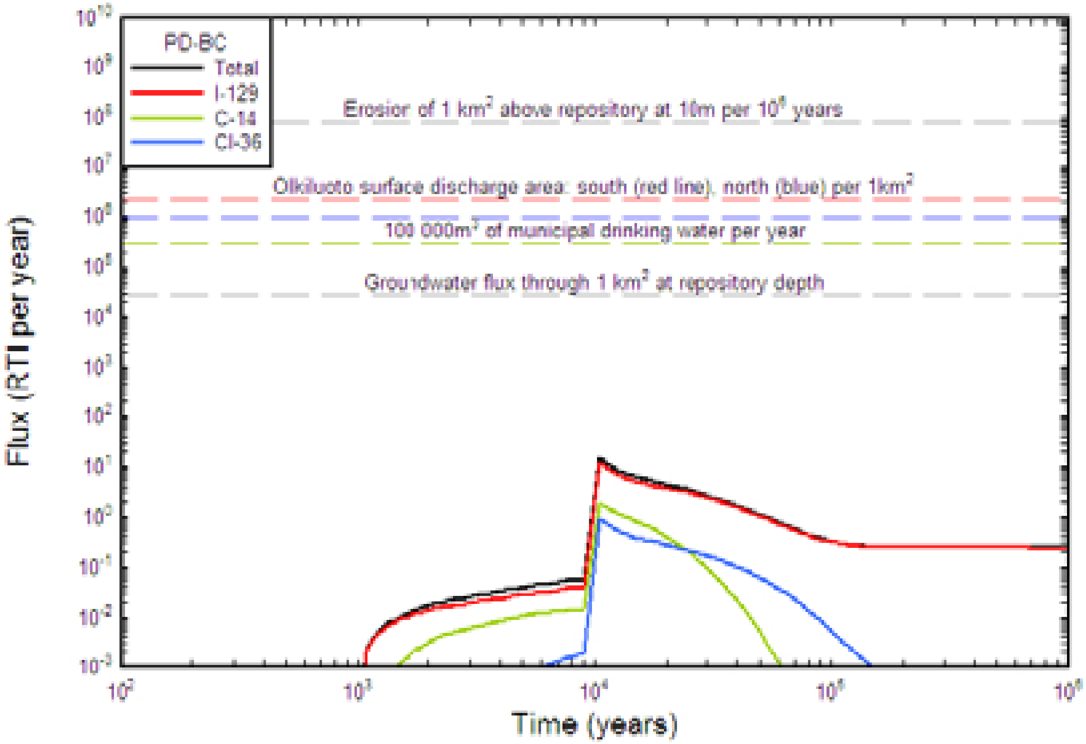

Figure 10-2. *Radiotoxicity flux from the KBS-3H repository into the biosphere for the penetrating defect base case (PD-BC) compared with a range of radiotoxicity fluxes due to naturally occurring radionuclides (see text and Appendix B, /Neall et al. 2007/ for further explanation).*

These evaluations confirm the insignificance of the calculated repository releases when compared with natural radiotoxicity fluxes associated with groundwater discharge in the Olkiluoto area, or erosion of the not-particularly uranium-rich rocks in the area.

10.5 Hazards due to chemotoxicity

Assessing the chemotoxicity of spent nuclear fuel and the releases from a spent fuel repository presents significant challenges due to paucity of data on the effects of relevant toxins on humans (as the application of data from experiments on animals is not straightforward) and because the diversity of toxins and their effects is much greater than for radioactivity /Chapman and McCombie 2003/. As a consequence, only a few radioactive waste management programmes (e.g. Andra in France, NIREX in the UK and OPG in Canada) have attempted to evaluate the chemical toxicity of selected elements or carried out a preliminary assessment for radioactive waste or spent fuel.

The study most relevant to the present safety assessment was carried out for Posiva by /Raiko and Nordman 1999/. This compared the concentrations of radionuclides and certain stable elements (e.g. copper and iron) released from a KBS-3V repository (based on the models and data used in TILA-99, /Vieno and Nordman 1999/) with available elemental drinking water standards to evaluate the overall safety. The concentrations of the selected elements were at least four to six orders of magnitude lower than the limits. Although the results are specific to TILA-99, the similarity of radionuclide release rates to the biosphere indicates that chemotoxicity is unlikely to be a significant issue for a KBS-3H repository.

10.6 Consequences of ultimate failure of the multi-barrier system in the farthest future

Over a sufficiently long time frame, one or more of the processes described in the previous sections will eventually lead to the failure of all canisters. For the majority of canisters, the most likely eventual failure mechanism is the slow corrosion of the copper shell, leading to failure after several hundred thousand years or more. In the hundreds to thousands of millions of years before the repository horizon is exposed at the surface by erosion and uplift, the evolution of the repository materials is uncertain and any comments are necessarily speculative. It is possible, but unlikely, that the fuel, radionuclides and repository construction materials will eventually become widely dispersed in the geological environment. It is more likely that at least some of the materials, including the spent fuel, will remain largely in situ. For example, the copper may be partly replaced under reducing conditions by a suite of copper sulphides, which are insoluble and not likely to become dispersed until erosion brings the repository horizon close to the surface, and the fuel matrix may experience only limited dissolution over time due to its relative geochemical stability. Thus, in some respects, after very long times the repository materials may tend to resemble a heterogeneous uranium ore body, perhaps analogous to granite- or sediment-hosted Cu-U deposits. The consequences of possible exhumation of the repository are difficult to assess, given the extreme length of time before this could occur. However, the processes involved are likely to be similar to exhumation of small uranium deposits where the local climatic and topographic conditions primarily determine the rate at which the ore body is dispersed.

11 Ongoing work and issues for further consideration

This chapter lists issues for further consideration that have been identified during the KBS-3H safety studies 2004–2007. Many of the issues below are relevant to both the KBS-3V and KBS-3H designs. Some of these issues are already the object of ongoing work and some are included in the next phase of the KBS-3H programme (2008-2009). The issues listed below are not prioritised and are to be considered in the context of the development of the general KBS-3 design, taking into account programmatic objectives and constraints, such as schedule and resources both in Posiva and SKB.

11.1 Methodological issues and limitations to be considered in future studies

Consistent with the "difference analysis" approach, at the start of the KBS-3H safety studies a decision was taken to follow the SR-Can approach for process selection and to accept the understanding and modelling basis presented in SR-Can in areas in which KBS-3H and KBS-3V are very similar, in particular in modelling canister and fuel processes. The reason for this is that major efforts would have to be made to advance the models beyond what was presented in SR-Can, and such advances were not part of the KBS-3H programme mandate. Such developments may, however, be considered in future project stages for both KBS-3H and 3V.

Although a broad range of assessment cases has been considered in the present safety assessment, the range of cases analysed is significantly smaller than that considered, for example, in either the TILA-99 or SR-Can safety assessments and not all conceivable uncertainties and combinations of uncertainties are covered. For example, uncertainties in the transport barrier provided by the geosphere, biosphere uncertainties and uncertainties related to future human actions are either not addressed or are analysed in less detail than others.

While use has been made, as far as possible, of well-tested and thoroughly reviewed models, computer codes and databases in analysing the assessment cases, each of these involves significant simplification of the real system. For example, significant limitations of the present safety assessment are acknowledged with the assumption of steady groundwater flow and composition (with the exception of one assessment case – PD-GMWV, see Chapter 9) and the highly simplified treatment of the geosphere fracture network in terms of a single representative fracture. Furthermore, data for analysing assessment cases are based on the preliminary information and design available at the time of writing the present report. The motivation for and plausibility (or conservatism) of selected parameter values and model assumptions used have been reported as much as possible given the time constraints in the supporting reports, and especially the Radionuclide Transport Report. This discussion is, however, often limited and largely qualitative. In the case of geosphere transport modelling, the modelling approach and parameter values used are based largely on TILA-99, and more recent developments in the understanding of the Olkiluoto site are used only to provide additional support for the parameter values selected (e.g. in terms of their conservatism). Uncertainties related to the geosphere transport barrier do not fall within the main focus of the present safety assessment, since most of the key differences between KBS-3V and KBS-3H repositories relate to the evolution and performance of the near field. A detailed analysis of the current understanding of the geosphere transport model parameters such as the flow and transport properties of the bedrock in terms of the transport resistance of the geosphere is, however, being undertaken in the context of the safety case for a KBS-3V repository at Olkiluoto. The use of stochastic discrete fracture network models (DFN) in geosphere modelling to better reflect the variability in groundwater flow is also being implemented.

In the biosphere part of the assessment, most of the ecosystem-specific models are largely unchanged compared with SR 97. However, the concept of landscape modelling, or chaining the ecosystems into a network, is rather new and has been previously applied only in SR-Can and in some case studies by Posiva. The studies have shown that, to track the fate and consequences of radionuclide releases to the biosphere, it is important to model the entire area of the biosphere – not just the first contaminated area affected by releases, but also areas downstream. Biosphere assessments conducted in this way are, however, rather cumbersome because of the large number of biosphere objects to be modeled. Thus, a tiered approach is envisaged for future safety assessments where first safety indicators like the WELL-2007 dose (and AgriWELL-2007 dose) are used, with detailed simulations carried out only for those few radionuclides contributing most to the dose.

The biosphere data applied in the present safety assessment are mainly the same data as those used previously in the SR 97 safety assessment, as well as in SR-Can, with only limited effort to ensure their consistency with the assumptions made for the near field and geosphere simulations in the present safety assessment. In the future, by concentrating on the most significant radionuclides, more resources could be spent on affirming the validity of the best estimate parameter values and distributions that contribute to the dose. Several tools to facilitate the process are already available, most importantly the Biosphere Assessment Database of Posiva with reviewing functions /Hjerpe 2006/, and the Eikos tool for sensitivity analyses /Ekström and Broed 2006/.

The issue of the hazard due to chemotoxicity is dealt with only briefly in Chapter 10 of the present report and in the Complementary Evaluations Report.

Finally, a comprehensive data report along the lines of that prepared for SR-Can /SKB 2006b/, with structured procedures for handling input data to radionuclide release and transport calculations, will be considered in any future safety studies for both the KBS-3V and KBS-3H repositories. Such a report will contribute to quality assurance and is likely to be required in support of a future safety case.

11.2 Issues related to the evolution of the engineered and natural barriers

This section describes remaining issues related to the evolution of the engineered and natural barriers to be considered in future safety studies, excluding issues of radionuclide release and transport, which are dealt with in Section 11.3. None of the following issues are specific to KBS-3H, although their significance to KBS-3V and KBS-3H may differ, for example due to the different nature and amounts of materials likely to be used in the two alternatives.

11.2.1 Evolution of conditions external to the repository

The Olkiluoto Site Description 2006 identifies a range of issues and uncertainties concerning the current site model, and discusses the activities being undertaken or proposed to address them (Chapter 10 in /Andersson et al. 2007/). There is a particular focus on groundwater flow and composition, the understanding of which is likely to improve significantly as characterisation and modelling of the Olkiluoto site continues. A specific issue for Olkiluoto site is the need for a better estimate on the rate of methane gas production and coupling of the methane gas production to the boundary conditions for other site models.

Specific areas of uncertainty particularly relevant to the discussions in the present safety assessment include:

- the long-term impact of backfilled and sealed exploration boreholes on groundwater flow (this issue is likely to be clarified as more specific information about the technical methods available to backfill and seal the boreholes becomes available),

- the possible effects on groundwater flow and composition of dissolved gases released from the deeper parts of the rock (methane and hydrogen),
- the impact of external conditions related to glaciations (e.g. taliks in permafrost, glacial melt water intrusion) on the long-term performance of the geosphere and the engineered barrier.

With regard to the impact of glaciations, additional information is needed to

- better understand rock-water interaction during the transient, high flow conditions associated with glaciation and the role of microbial reactions under glacial and post-glacial conditions,
- determine whether glacial meltwater (and possibly oxygen) penetrated to repository depth during past glaciations,
- provide insight as to the hydrogeology at Olkiluoto during permafrost/glacial conditions,
- better understand the hydromechanical effects of successive glaciations.

Some potentially significant hydromechanical effects include hydraulic jacking as well as the existence of a highly anisotropic stress field in relation to deglaciation that could possibly open fractures. Modelling work on these issues is underway in Sweden.

A further issue that is particularly relevant if the onset of any future change to glacial conditions is delayed by anthropogenic emissions is the likely reduction in ionic strength of the groundwater during a prolonged period of isostatic uplift. Significant (though uncertain) buffer erosion could occur if the ionic strength were to fall below the Critical Coagulation Concentration (CCC). On the basis of current shallow groundwater composition, it is judged unlikely that this would occur (other than in association with glaciation, which is discussed in Section 7.2.6) because of the interaction of this water with the rock and fracture-filling materials, but further information is needed.

11.2.2 Early evolution of the buffer

The evolution of the buffer, including the possibility of erosion by transient water flows (piping) during operations and subsequent saturation, drying/wetting, impact of iron saturation and cementation due to silica precipitation are all issues requiring more thorough investigation. Limitation of piping and erosion is discussed further in the context of design issues (Section 11.4.1).

Another issue for further consideration is the effect of the various strain mechanisms that are involved in the early (and long-term) evolution of the supercontainer shells on the outer part of the buffer. Depending on how the shell deforms during saturation, it may generate heterogeneities in the buffer.

11.2.3 Iron/bentonite interaction

Iron/bentonite interaction and the possible alteration of montmorillonite and formation of new minerals are potentially detrimental to the safety functions of the buffer and subject to significant uncertainties. In the study by /Carlson et al. 2006/, for example, a substantial increase in hydraulic conductivity was found in some of the samples which had been in contact with iron for a period of up to 3 years. The swelling pressure was, however, rather unaffected in these systems, indicating inhomogeneity, with localised high density/low hydraulic conductivity volumes and low density/high hydraulic conductivity volumes. Cementing was probably one reason for the localised effect. This remains, however, an issue for further study.

There is also significant uncertainty regarding the possible reduction of structural iron in smectites from Fe(III) to Fe(II) due to iron-clay interaction, since available experimental data are not representative of repository conditions. The key issue for the physical performance of the buffer is the structural change that this might cause. Reduction of Fe(III) to Fe(II) could, in principle, have consequences for the stability and to some extent the swelling pressure and hydraulic properties of the buffer.

Results of preliminary 1-D reactive transport modelling reported in /Wersin et al. 2007/ indicate that the extent of the zone potentially undergoing mineral transformation due to iron-clay interaction is likely to remain spatially limited (a few centimetres) for very long times. Nevertheless, in view of its potential impact on mass transfer at the buffer/rock interface, the development of a more realistic model of the alteration front remains an issue for further study). Recommendations for continuing studies of iron/bentonite interaction are given in /Wersin et al. 2007/. These include, from the experimental side, long-term iron/bentonite interaction studies that include measurements of physical properties (swelling pressure, hydraulic conductivity) and well-controlled diffusion experiments with Fe(II). Studies to identify the reducing species that may cause electron transfer to the structural Fe(III) and to elucidate the mechanism involved are being undertaken by SKB. The processes related to the interaction of iron with the buffer are complex and the consequent impact on the required properties of bentonite is far from being completely understood.

Overall, more effort is required in future studies to elucidate chemical changes, and particularly cementation effects, and the processes involved in iron-clay interaction. The use of natural analogues to support these studies is likely to be considered. The possible use of alternative supercontainer materials to circumvent the potentially adverse effects of iron-bentonite interaction is mentioned briefly in Section 11.4.

11.2.4 Gas pressurisation, migration and the properties of the EDZ

The corrosion of the supercontainer shells and the other steel components of a KBS-3H repository will give rise to significant gas pressures in tight drift sections, where the gas cannot readily escape into intersecting transmissive fractures in the rock. The EDZ is a potential migration route for gas away from tight drift sections, but the properties of the EDZ, and thus the magnitude of the gas pressures that will arise in these sections, are subject to uncertainties.

Following canister failure, the corrosion of the cast iron canister insert will also lead to significant gas generation. It is generally assumed in the evaluation of assessment cases that gas generated inside the canisters, principally hydrogen from the corrosion of the insert, has no impact on the transport properties of the buffer. However, while the effect of gas and its transport through a saturated and chemically unperturbed bentonite clay buffer are reasonably well known, there is little understanding of how Fe-saturated or otherwise chemically altered smectite clay transports gas. It is known, however, that the microstructural constitution of Fe-saturated MX-80 is characterised by channels, which could result in gas breakthrough at relatively low pressures. Gas flow could potentially cause erosion and widening of the channels. The self-sealing capability of altered clay is also currently unknown.

11.2.5 Microbial activity

There is a potential for microbial activity at the boundary between sulphate-rich water and the deeper, methane zone at the Olkiluoto site, giving rise to some uncertainty in the evolution of sulphide concentrations in the groundwater. The hydrogen generated during the first thousands of years due to corrosion of the supercontainer and other steel structural components may also affect the corrosion of the canisters via its effect on the microbial reduction of sulphate to sulphide, but the sulphides formed may be precipitated as iron sulphides by reacting with the iron corrosion products, thus reducing the flux of sulphide to the canister surface. Although the impact on canister lifetime is expected to be limited because of the slow migration of microbially-produced sulphide through the bulk of the buffer, this issue needs to be further evaluated, especially with respect to the transport rate of methane and the kinetics of sulphate reduction in the rock.

Within the bulk of the buffer, microbial activity is expected to be suppressed and decrease with time due to the increasing buffer swelling pressure. The criterion currently set for negligible microbial activity to be assumed is that the swelling pressure should exceed 2 MPa. The highest value of buffer density at which significant microbial activity can occur in compacted bentonite is an issue currently under investigation by SKB.

11.2.6 Effect of hydrogen gas on porewater chemistry

It is generally assumed that there is no significant interaction between hydrogen and minerals in the buffer or with porewater. This assumption, however, still awaits experimental verification. The presence of high hydrogen partial pressures due to the corrosion of the supercontainer shells and the other steel components of a KBS-3H repository may have an impact on the bentonite porewater chemistry. This potential impact has not yet been fully evaluated. Various factors need to be considered, including acid-base equilibria and the pH buffering capacity of bentonite, as well as the limited timeframe of hydrogen production of several thousand years. The overall impacts, in particular any effects on the near-field conditions, should be considered in future studies.

11.2.7 Interactions involving cement and stray materials

Interactions involving cement and stray materials are subject to significant uncertainties, including lack of knowledge of the general mechanism of cement/bentonite interaction. Simple mass balance calculations on the impact of alkaline leachates on bentonite have been performed, but these excluded a number of phenomena, such as the precipitation/dissolution of secondary minerals in bentonite. Although the limited amounts of alkaline leachates from low-pH cement in the drift are not expected to have significant adverse effects on the overall safety functions of the canister, buffer or host rock, this is likely to be the subject of further studies. Currently, some adverse effects on the mass transport properties of the buffer/rock interface cannot be excluded. The effects of highly alkaline conditions on the solubilities of key elements also need to be assessed, although progress is being made in other programmes dealing with cementitious systems for TRU disposal which are collecting thermodynamic data for these elements at high pH conditions. Finally, studies are ongoing to assess the impact of cement additives, e.g. superplasticisers, on radionuclide transport in the event of canister failure.

11.2.8 Thermally-induced rock spalling

Ongoing work by Posiva and SKB aims to improve understanding of rock spalling processes, and to find means to reduce or prevent spalling if necessary. For example, the possibility of excavation-induced rock spalling is being further studied via the Prediction-Outcome studies currently underway in Posiva's ONKALO access ramp /Andersson et al. 2007/. Prior to the emplacement of spent fuel, alignment of the deposition drifts with the direction of the principal stress in the rock should ensure that little or no spalling will occur. However, the results of these studies should also provide a firmer basis for predictions of which drift sections may undergo thermally-induced rock spalling subsequent to spent fuel emplacement. This issue is of particular concern in the tightest drift sections, where buffer swelling pressure on the drift wall takes longest to develop. Measures to reduce or prevent thermally-induced rock spalling are also mentioned in the context of design issues in Section 11.4.2.

11.2.9 Buffer erosion by dilute glacial meltwater

There is currently no adequate quantitative understanding of chemical erosion of the buffer following the potential penetration of dilute glacial meltwater to repository depth. The process is modelled in SR-Can, and is one of the key issues identified that could cause significant loss of buffer and consequent corrosion and failure of a number of canisters. It is, however, acknowledged that better understanding of the erosion process could lead to models that yield either lower or higher buffer loss rates, and a new model is currently under development by SKB. More generally, the overall effect of chemical erosion on the buffer during successive glaciations is an issue currently being investigated at SKB and Posiva, and needs also to be addressed in future KBS-3H safety studies. The related issue of colloid-facilitated radionuclide transport is mentioned in Section 11.3.8.

11.3 Issues related to radionuclide release and transport

This section describes remaining issues related to radionuclide release and transport modelling. Again, none of the following issues is specific to KBS-3H.

11.3.1 Radionuclide inventory

The partitioning of radionuclide inventories between the fuel matrix, instant release fraction, Zircaloy and other metal parts is uncertain; the values used in the present safety assessment are based on expert judgement, guided by considerations in /Anttila 2005a/ and /Werme et al. 2004/. Expert judgment is particularly applied regarding the distribution between the IRF and the fuel matrix as outlined in /Werme at al 2004/. In this case, uncertainty is handled by providing distributions of values from various measurements which include a pessimistic upper limit. Although it is difficult to improve the knowledge of the distribution of radionuclides between the IRF and within the fuel rods (which depends e.g. on reactor types, fuel manufacturer, spatial configuration in the reactor) confidence may be improved by studying fuel types at high burnups for which there are few data available.

11.3.2 Probability of canister failure

Three different canister failure modes are judged to be plausible in a million year time frame for a KBS-3H (or KBS-3V) repository at Olkiluoto: (i), the presence of an initial, penetrating defect in the copper shell, (ii), canister failure by corrosion following perturbation of the buffer or buffer/rock interface and (iii), canister failure due to secondary shear movements on rock fractures in the event of a large earthquake.

Posiva is not yet taking any position on the likelihood of occurrence of canisters with initial penetrating defects. A qualification programme for the applicable examination procedures for Finnish canister component manufacture and sealing will be developed by the end of 2009 and executed by the end of 2012, before the application for the encapsulation plant construction license is submitted.

Canister failure by corrosion in a million year time frame is judged to be possible only if there is significant perturbation of the buffer or buffer/rock interface, for example due to chemical erosion by dilute glacial meltwater, allowing an increased rate of migration of sulphide from the groundwater to the canister surface. As noted in Section 11.2.9, however, there is no adequate quantitative understanding of chemical erosion on which to base a reliable estimate of the rate of canister failure by this mechanism. As also noted in Section 11.2.5, there is uncertainty in the evolution of sulphide concentrations in the groundwater, which is associated with the potential for microbial activity at the boundary between sulphate rich water and the deeper, methane zone at the Olkiluoto site.

Only in the case of canister failure by rock shear has an estimate been made of the number of canisters potentially affected. This estimate, however, has some significant limitations, being based, for example, on fracture network data that have since been updated. Furthermore, the cumulative effect of the impact of many earthquakes over a prolonged period on the buffer and canisters is an issue for further investigation. Finally, methods to identify and avoid potentially problematic fractures have not, so far, been evaluated in detail (see Section 11.4.4).

In the present safety assessment, each assessment case postulates a single canister failure by one or other of the above failure modes at a given time, and the resultant releases of radionuclides to the biosphere are evaluated. The likelihood or frequency of canister failure by each mode must, however, be evaluated in any future safety case.

11.3.3 Internal evolution of a failed canister

There are many uncertainties in the internal evolution of a failed canister, most of which can be treated in safety assessment calculations, e.g. using conservative assumptions. It is possible that a better understanding of, for example, processes leading to water ingress, fuel matrix dissolution and the establishment of a transport pathway from the canister interior to the buffer could allow some of these uncertainties to be reduced.

11.3.4 Solubility limitation and speciation

Solubility limits, which are applied in the present safety assessment inside a failed canister and at the buffer/rock interface, are subject to a number of significant uncertainties.

One important issue related to solubility limits is element speciation. For some elements, including carbon, the chemical form is currently not precisely known. A few elements (e.g. niobium, plutonium, thorium and uranium) are thought to be present in anionic, neutral, or cationic forms depending on the groundwater composition, and especially on pH and redox conditions and concentration of dissolved carbonate in the groundwater. This is an issue, therefore, that is dependent on both the knowledge of the correct groundwater composition and accuracy of available thermodynamic data.

In the present safety assessment, the transport of one dominant chemical species is considered for each element, whereas, in reality, there may be a mixture of different species, each with different transport properties. The species that are determined to be the most abundant in the repository near field following release to solution are, in most cases, the ones for which transport calculations are performed. It is, however, possible that some of the less abundant species will be the more mobile and so, in reality, will be released at greater rates to the biosphere. Thus, in some uncertain cases, e.g. thorium, the dominant species is selected conservatively based on the known sorption properties of the element, rather than on abundance. The kinetics of transformation of chemical species during transport is also uncertain. These are issues that will require consideration in future safety assessments.

Solubility limits used in the evaluation of assessment cases are based on a set of reference waters, corresponding to different possible groundwater compositions. In reality, however, groundwater composition, and, therefore, solubilities, will vary in space and time. The salinity of the water at Olkiluoto site during different periods has been modelled in the 3V Evolution Report /Pastina and Hellä 2006/, and the impact of the time-dependence of groundwater components of importance in the estimation of solubility limits will be considered in the coming 3V studies.

For a given reference water, uncertainties in solubility limits arise from the fact that the concentrations are not known or the concentrations are associated with large uncertainties for some important groundwater components, such as phosphates and organic acids. The lack of data on phosphate concentration (or the fact that phosphate concentration is below the detection limit) has an important impact on the solubility limiting phases and the solubility limits. Organic compounds and formation of organic complexes have not been considered in the assessment of solubilities and speciation of the elements. Uncertainties also arise due to the limited overall understanding and quantification of the redox- and pH-buffering capacity of the rock during different periods. Furthermore, some of the thermodynamic data for the radionuclides of interest are still associated with significant uncertainties, as exemplified by the recent data published for Ni /Gamsjäger et al. 2005/.

Finally, the possibility of precipitation or co-precipitation at the boundary between the buffer and rock, leading to colloid formation and migration in the geosphere, has not been taken into account in the present study and is an issue for future studies.

11.3.5 Sorption

There are a number of uncertainties affecting the sorption values used in the present safety assessment. There is a general lack of experimental data for some elements under relevant conditions, in particular Nb and Mo. In the near field, organic substances or their degradation products could, for example, form complexes with radionuclides, which would lower radionuclide sorption and thus increase radionuclide transport rates. Another source of uncertainty is the impact of Fe(II) from the corrosion of the insert of a failed canister on the capacity of the buffer to sorb some radionuclides. Fe(II) could, in particular, compete with the sorption of species such as Ni(II) and Sr(II), and weaken the barrier function of the buffer with respect to Ni and Sr. In the present safety assessment, however, it is argued on the basis of scoping calculations that the impact of such uncertainties on overall release rates to the biosphere is limited. Iron corrosion products may not only perturb the sorption properties of the buffer, but may also themselves sorb radionuclides. Although this process is favourable to safety, it is also possible that changes in geochemical conditions and/or ageing and transformation of the surfaces of the corrosion products could release sorbed radionuclides. This is also particularly relevant in the case of Ni(II).

Overall, there is a need to update geosphere parameter values based on more recent information on the host rock at Olkiluoto, including the changes caused by the evolution of groundwater chemistry. These need to be considered in parallel with improvements in knowledge regarding the speciation of different elements, including an assessment of the role of organic compounds and formation of colloids.

11.3.6 Buffer transport properties

In has been assumed in radionuclide release and transport calculations that transport properties throughout most of the buffer correspond to those of saturated MX-80 bentonite. The basic properties of the MX-80 bentonite, including transport properties, are fairly well known based on about 30 years of R&D. In reality, the bentonite can undergo a variety of changes during its early evolution, especially in drift sections where saturation of the buffer is very slow, as discussed in Section 11.2. A better understanding of the changing transport properties of the buffer over time is required, taking into account its evolution from virgin plastic clay to clay that is potentially altered, for example by the presence of iron, precipitated silica and stray materials.

11.3.7 Treatment of the buffer/rock interface

Various features and processes have been identified in the safety assessment that may lead to detrimental changes in the mass transfer properties of the buffer/rock interface (e.g. iron/bentonite interaction, thermally-induced rock spalling). In considering the effects of such changes on radionuclide release and transport, the approach taken is to treat the perturbed zone as a mixing tank, and to evaluate groundwater flow through this mixing tank based on the highly conservative assumption that it has a high (effectively infinite) hydraulic conductivity. In reality, even if the perturbed zone has a hydraulic conductivity that is higher than either the rock matrix or the unperturbed buffer, hydraulic conductivity may remain sufficiently low to limit the flow of water through the zone. It is even possible that a buffer transformed, for example, by iron/bentonite interaction could be impermeable and thus continue to limit mass transfer across the buffer/rock interface significantly. Determining some bounding estimates for the hydraulic properties of the perturbed buffer/rock interface is an issue for future studies.

11.3.8 Geosphere transport properties and processes

In the present safety assessment, the dominant radionuclide transport mechanism in the geosphere is assumed to be advection of dissolved radionuclides in flowing groundwater, retarded by matrix diffusion and sorption. Steady groundwater flow and transport properties that are constant in space and time are assumed. These assumptions are not well supported. For example, the possible impact of groundwater or repository-derived colloids on radionuclide transport has not been addressed in the present safety assessment[31]. The potential impact of high-pH leachates on the transport properties of the geosphere has also not been evaluated. Assumptions such as the negligible impact of colloids and of high-pH leachates on geosphere transport are made to simplify the present assessment, the focus of which is on the repository near field, since this is where most of the differences between KBS-3H and KBS-3V arise. As noted in Section 11.1, a detailed analysis of the current understanding of the geosphere transport model parameters is being undertaken in the context of the safety case for a KBS-3V repository at Olkiluoto, and the findings should also be applicable to KBS-3H.

Concerning the impact of earthquakes, the possibility of detecting fractures with the potential to undergo shear movements of 0.1 m or greater in the event of a large earthquake, and of avoiding such fractures during repository construction, has not yet been evaluated in detail. Furthermore, the cumulative effect of the impact of many earthquakes over a prolonged period on the buffer and canisters is an issue for further investigation.

11.3.9 Biosphere transport properties and processes

In the present safety assessment, the transport in the biosphere of all radionuclides, with the exception of for radiocarbon, is treated using the novel approach of landscape modelling. Landscape modelling takes account of radionuclide transport between different ecosystems and ensures that radionuclide inventories are properly conserved when the landscape evolves with time. In order to treat the special case of radiocarbon, a new model has been developed and implemented, based on a modified specific activity approach.

An issue not addressed thoroughly in the present biosphere analyses /Broed et al. 2007/ is the spatial distribution of the radionuclide release from the geosphere to the biosphere and associated uncertainties; a study addressing this issue is currently being undertaken. A related issue also requiring further investigation is the forecasting of future surface hydrology. It is foreseen that the evolution of surface hydrology will be modelled by extending the existing model for the present-day surface hydrology at Olkiluoto into the future. Input will be provided by terrain forecasts (e.g. calculated shoreline displacement), which will be translated, or interpreted into the boundary conditions required for the modelling of the groundwater flow in the bedrock. Surface hydrological modelling will also be used to better determine the locations of radionuclide release to the biosphere and as a basis for improving the transport modelling of any radionuclides that are released through the overburden.

11.4 Design issues

The present safety assessment was conducted while design development and associated laboratory testing was still underway. It was necessary to select a reference design for the assessment (the Basic Design), even though there were uncertainties regarding the feasibility of implementing this preliminary design in practice. The feasibility of implementing a given design is assumed in safety assessment, and this assumption must be justified as part of any future safety case. Ongoing design development that should improve the robustness of any future safety case are also summarised below.

[31] In SR-Can it was concluded that colloid-facilitated radionuclide transport did not need to be included in the consequence calculations, with the possible exception of glacial conditions (Section 10.4.2 of /SKB 2006a/). Therefore, a bounding case where geosphere retention was neglected for glacial conditions was evaluated. However, no evaluation of the potential significance of colloid-facilitated transport at the Olkiluoto site has been made in the present safety assessment.

11.4.1 Avoidance of distance block displacement and deformation during early evolution

In the current reference design, the initial gaps between the distance blocks and adjoining supercontainers are made small. If these gaps remain small, then, when the void space around a supercontainer becomes water filled and the water pressure increases towards the hydrostatic pressure at repository depth, this pressure is exerted only on narrow annuli around the outer perimeters of the vertical faces of the adjoining distance blocks, rather than on the full surface areas of the faces. This is important since, if void space around a neighbouring supercontainer section is more slowly filled, then the resulting pressure difference across the separating distance block could result in displacement of this distance block relative to the supercontainer. This could increase the likelihood of piping along the distance block/host rock interface, and would affect the final saturated density of the buffer in a manner that is not possible to predict.

Maintaining a small gap between the distance blocks and adjoining supercontainers is therefore critical in the current reference design. In this design, steel fixing rings will be installed, where necessary, to avoid displacement of the distance blocks prior to the installation of compartment and drift end plugs. It has, however, recently been shown in modelling studies reported in the Design Description 2006 /Autio et al. 2007/ that deformation of the blocks may still occur due to hydraulic pressure differences along the drift, which could lead to the adverse consequences described above. Design alternatives that are less sensitive to these phenomena are being developed, including the design variant termed DAWE (Drainage, Artificial Watering and air Evacuation). These designs and the situations in which they might be implemented are likely to be further studied as part of the repository design work.

11.4.2 Avoidance or limitation of thermally induced rock spalling

In the current reference design, significant thermally induced spalling could occur on a timescale of a few years in relatively tight drift sections (wall rock hydraulic conductivity less than about $10^{-12}\ m\ s^{-1}$). However, there are indications that pressures much smaller than the full buffer swelling pressure are likely to be sufficient to suppress spalling (/Chou et al. 2002/, and the APSE experiment at the Äspö hard rock laboratory, reported in /Andersson and Eng 2005/). The pressure required is thought to be in the order of 150 to 200 kPa, but is uncertain, and is likely to require further investigation. The use of pellets to prevent spalling is being studied in the context of the KBS-3V design, and such measures may also be considered in future KBS-3H design studies.

11.4.3 Possible alternative supercontainer materials

A concern with the use of steel in the supercontainer shell and some other engineered structures in the drift is its possible detrimental impact on the physical properties of buffer (Section 11.2.3). Only few relevant experimental studies are currently available and further studies to address the processes involved and their impact on, for example, the swelling and hydraulic properties of bentonite are being initiated. The findings of these studies will be balanced against the favourable qualities of steel, such as the scavenging effects of its corrosion products on sulphide and oxygen, in deciding whether or not steel remains a suitable candidate material for the supercontainer shell.

As a complement to these studies, materials such as titanium are being studied as possible alternatives to steel, since they avoid the complex evolution and uncertainties associated with iron/bentonite interaction at the buffer/rock interface. A preliminary evaluation of the safety implications of using alternative materials has been made /SKB/Posiva 2008/ and the conclusion is that titanium is expected to have a very low reactivity with bentonite and – the rate of corrosion is much lower. Nevertheless, the long-term safety implications of these materials, such as their potential impact on the safety functions of the buffer, still need to be more thoroughly addressed.

11.4.4 Layout to avoid potentially problematic fractures

Shear movements on rock fractures sufficient to damage the canisters in the event of a large earthquake are only possible on fractures above a certain size. In SR-Can, the calculated probability of canister failure is significantly reduced by assuming that the Expanded Full Perimeter Criterion (EFPC) is applied, whereby large fractures intersecting both the full perimeter of a KBS-3V deposition tunnel and the deposition hole are assumed to be readily observable and avoided /Munier 2006/. It is uncertain, however, whether or not a similar line of reasoning can be applied to a KBS-3H repository, without rendering a large proportion of the drift unusable. Thus, no criterion similar to the EFPC is applied in the calculation of the likelihood of canister damage, although this remains an issue for further study.

12 Conclusions

This chapter presents conclusions from the long-term safety assessment carried out for a preliminary design of a KBS-3H repository at the Olkiluoto site.

The conclusions are as follows:

1. In the absence of any initial penetrating defect in the canisters, no canister failures should occur during the first several thousand years after canister deposition provided the repository system evolves as expected. Thereafter, the processes that are potentially the most detrimental to repository safety are related to glacial conditions. This was also a main conclusion arising from SR-Can in the case of a KBS-3V repository for spent fuel at two Swedish sites.
2. Safety issues related to a future change to glacial conditions at the Olkiluoto site are generally the same as those identified in SR-Can for the KBS-3V design at Swedish sites, the most significant being canister failure due to rock shear in the event of a large, post-glacial earthquake and loss of buffer from exposure to glacial meltwater, which may lead to early failure of some canisters by corrosion. There are, however, some differences compared with SR-Can and KBS-3V, e.g. the probability of, and possibility of avoiding by design, fractures that can undergo rock shear movements that damage canisters in the event of a large post-glacial earthquake. Furthermore, in the case of KBS-3H, loss of buffer around one canister due to exposure to glacial meltwater may affect the corrosion rate of neighbouring canisters, since the buffer density along the drift will tend to homogenise over time. This also means that the impact on buffer density and on the corrosion rate of the first canister will diminish with time. In the case of KBS-3V, on the other hand, buffer loss around one canister will not affect the state of the buffer around the other canisters.
3. A difference analysis has shown that the key differences in the evolution and performance of the KBS-3H and KBS-3V designs relate mainly to the engineered barrier system and to the impact of local variations in the rate of groundwater inflow on buffer saturation along the drifts. The safety functions of the geosphere are generally not expected to differ significantly between the two designs, but the importance of some geosphere properties may differ, e.g. the KBS-3H design is more sensitive to sub-vertical fractures with respect to potential damage to the engineered barrier system by rock shear.
4. No features or processes that are specific to KBS-3H have been identified that could lead to a loss or substantial degradation of the safety functions of the engineered barrier system over a million year time frame. However, the degree to which fractures with the potential to undergo shear movements that damage the engineered barriers in the event of a large earthquake can be identified and avoided remains to be evaluated, and may be different for KBS-3H compared with KBS-3V.
5. Particularly in tight drift sections, the gas generated by the steel components of the KBS-3H repository external to the canister in the current reference design (principally the supercontainer shell) may accumulate at the buffer/rock interface, possibly resulting in a prolonged period during which significant inflow of water from the surrounding rock will be limited and the buffer will remain only partially saturated.
6. The timing of eventual canister failure by corrosion may be affected by perturbations to the buffer/rock interface caused, for example, by the presence of the steel supercontainer shell and its corrosion products. The issues related to the impact of iron and its corrosion products on the buffer bentonite are potentially detrimental to the safety functions of the buffer and subject to significant uncertainties. Hydrogen generation during the first thousands of year may also affect the corrosion of the canisters via its effect on the microbial reduction of sulphate to sulphide, but the sulphides formed may be precipitated as iron sulphides by reacting

with the iron corrosion products, thus reducing the flux of sulphide to the canister surface. The conclusions from the analyses performed are that these perturbations are not expected to lead to canister failure by corrosion within a million year time frame.

7. Radionuclide release from the repository near field in the event of canister failure may also be affected by perturbations to the buffer/rock interface, but in all cases releases are limited and comply with Finnish regulatory criteria. Only single canister failure cases have, however, been considered and the possibility of multiple canister failures must be addressed in future studies.

8. Several issues have been identified for further study, many of which are relevant to both KBS-3V and KBS-3H. These include, for example, site-specific issues such as the transport rate of abiogenic methane and the kinetics of sulphate reduction in the rock. While some issues, such as those related to gas generation prior to canister failure, are relevant mainly to KBS-3H, it should also be noted that there are some issues that are specific to KBS-3V (see Table 3-1).

These conclusions are based on the analysis of a KBS-3H reference design, termed the Basic Design and its application to the Olkiluoto site. It should, however, be emphasised that this choice of reference design is preliminary, and that design alternatives are being developed. Furthermore, differences between the fuel, canisters and repository sites under consideration in Sweden and Finland will have to be considered in transferring the detailed findings of the present safety assessment to a Swedish context. On the other hand, the focus of the safety assessment is on the evolution and performance of the engineered barrier system, and, with the exception of overall inventory, this system is broadly similar in the Swedish and Finnish contexts although local variations e.g., in the hydraulic conditions will have an impact on the evolution of the system. Still, many of the broad findings on the engineered barrier system are expected to be readily transferable.

The present safety assessment has some important limitations. In particular, the analysis of a limited range of assessment cases with highly simplified models, especially of the geosphere, is not considered sufficient to test whether a KBS-3H repository at the Olkiluoto site satisfies all relevant regulatory guidelines. In SR-Can, for example, the treatment of geosphere variability had an important impact on the conclusions, affecting, for example, the likelihood of canister failure by corrosion and this would also be expected to be the case for a KBS-3H repository at Olkiluoto.

Another limitation is that the feasibility of implementing the current reference design has been assumed, even though several design issues remain to be addressed. A design alternative termed DAWE is being developed to address some of concerns regarding the feasibility of implementing the Basic Design. Its evolution has been assessed, though only to a limited extent, as described in the Process Report. The evolution of the buffer prior to buffer saturation in the DAWE design is likely to require particular attention in future studies. However, provided a DAWE repository can be shown to reach a state of buffer saturation without any significant detrimental perturbations to the main barriers, it should thereafter evolve in a similar manner to the Basic Design and the same conclusions regarding long-term safety should apply to both designs. Whatever the final chosen design may be, feasibility of implementation must be justified as part of any future safety case.

Finally, the long-term safety implications of variability or errors in manufacturing or installation of system components have also not been systematically addressed in the present safety assessment, although the impact of an initial penetrating defect in a canister has been considered in radionuclide release and transport calculations.

In spite of these limitations, it can be concluded based on the present safety assessment that the KBS-3H design alternative offers potential for the full demonstration of safety for a repository at Olkiluoto site and for the demonstration that it fulfills the same long-term safety requirements as KBS-3V. A full safety assessment of the KBS-3H design, including a more detailed treatment of geosphere variability and a full consideration of design feasibility and the effects of variability or errors in manufacturing or installation of system components, is not likely to be made until such time as an updated KBS-3V safety assessment of the Olkiluoto site is available (i.e. in 2012). Nevertheless, studies are already being undertaken to address critical scientific and design issues. These include the further development of DAWE to avoid the possibility of distance block displacement or deformation, which could lead to significant piping and erosion, as well as studies of iron/bentonite interaction and the possible use of alternative materials such as titanium for the supercontainer shell and some other engineered structures in the drift.

References

Ahokas H, Hellä P, Ahokas T, Hansen J, Koskinen K, Lehtinen A, Koskinen L, Löfman J, Mészaros F, Partamies S, Pitkänen P, Sievänen U, Marcos N, Snellman M, Vieno T, 2006. Control of water inflow and use of cement in ONKALO after penetration of fracture zone R19. Posiva Working Report 2006-45. Posiva Oy, Olkiluoto, Finland.

Ahonen L, Vieno T, 1994. Effects of glacial meltwater on corrosion of copper canisters. Report YJT-95-19. Nuclear Waste Commission of Finnish Power Companies (YJT), Helsinki, Finland.

Andersson J, 1999. SR 97 Data and data uncertainties. SKB TR-99-09, Svensk Kärnbränslehantering AB.

Andersson C, Eng A, 2005. Äspö pillar stability experiment. Final experimental design, monitoring results and observations. SKB R-05-02, Svensk Kärnbränslehantering AB.

Andersson J, Ahokas H, Hudson J A, Koskinen L, Luukkonen A, Löfman J, Keto V, Pitkänen P, Mattila J, Ikonen A T K, Ylä-Mella M, 2007. Olkiluoto Site Description. POSIVA 2007-03. Posiva Oy, Olkiluoto, Finland.

Anttila P, Ahokas H, Front K, Heikkinen E, Hinkkanen H, Johansson E, Paulamäki S, Riekkola R, Saari J, Saksa P, Snellman M, Wikström L, Öhberg A, 1999. Final disposal of spent nuclear fuel in Finnish bedrock – Olkiluoto site report. POSIVA 99-10. Posiva Oy, Helsinki, Finland.

Anttila M, 2005a. Radioactive characteristics of the spent fuel of the Finnish nuclear power plants. Posiva Working Report 2005-71. Posiva Oy, Olkiluoto, Finland.

Anttila M, 2005b. Criticality safety calculations for three types of final disposal canisters. Posiva Working Report 2005-13. Posiva Oy, Olkiluoto, Finland.

Autio J, Hjerpe T, Siitari-Kauppi M, 2003. Porosity, Diffusivity and Permeability of EDZ in Crystalline Rock and Effect on the Migration in a KBS-3 Type repository. Proceedings of a European Commission Cluster conference and workshop on Impact of the excavation disturbed or damaged zone (EDZ) on the performance of radioactive waste geological repositories, Luxemburg, 3 to 5 November 2003, EUR 21028 EN, p. 149–155.

Autio J, 2007. KBS-3H Design Description 2005. Posiva Working Report 2007-11 and SKB R-08-29. Posiva Oy, Olkiluoto, Finland and Svensk Kärnbränslehantering AB, Sweden.

Autio J, Börgesson L, Sandén T, Rönnqvist P-E, Berghäll J, Kotola R, Parkkinen I, Johansson E, Hagros A, Eriksson M, 2007. KBS-3H Design Description 2006. Posiva Working Report 2007-107 and SKB R-08-32. Posiva Oy, Olkiluoto, Finland and Svensk Kärnbränslehantering AB, Sweden.

Autio J, Anttila P, Börgesson L, Sandén T, Rönnqvist P-E, Johansson E, Hagros A, Eriksson M, Halvarson B, Berghäll J, Kotola R, Parkkinen I, 2008. KBS-3H Design Description 2007, POSIVA 2008-01 (in preparation). Posiva Oy, Olkiluoto, Finland and SKB R-08-44. Posiva Oy, Olkiluoto, Finland and Svensk Kärnbränslehantering AB, Sweden.

Avila R, Bergström U, 2006. Methodology for calculation of doses to man and implementation in PANDORA. Posiva Working Report 2006-56 and SKB R-06-68, Posiva Oy, Olkiluoto, Finland and Svensk Kärnbränslehantering AB, Sweden.

Avila R, Pröhl G, 2007. Models for assessment of doses from underground releases of C-14 Posiva Working Report 2007-107. Posiva Oy, Olkiluoto, Finland.

Blomqvist R, Ruskeeniemi T, Kaija J, Paananen M, Smellie J, Grundfeldt B, Pederson K, Bruno J, Pérez del Villar L, Cera E, Rasilainen K, Pitkänen P, Suksi J, Casanova J, Read D, Frape S, 2000. The Palmottu natural analogue project. Phase II. Transport of radionuclides in a natural flow system at palmottu. Final report. CEC Nuclear Science and Technology Series EUR 19611 EN. CEC, Luxembourg.

Broed R, 2007. Landscape model configuration for biosphere analysis of selected cases in TILA-99 and in KBS-3H safety evaluation, 2007. Posiva Working Report 2007-108. Posiva Oy, Olkiluoto, Finland.

Broed R, Avila R, Bergström U, Hjerpe T, Ikonen A T K, 2007. Biosphere analysis for selected cases in TILA-99 and in KBS-3H safety evaluation, 2007. Posiva Working Report 2007-109. Posiva Oy, Olkiluoto, Finland.

Brookins D G, 1990. Radionuclide behaviour at the Oklo nuclear reactor, Gabon. Waste Management, 10, 285–296.

Brydsten L, 2004. A mathematical model for lake ontogeny in terms of filling with sediments and macrophyte vegetation. SKB TR-04-09, Svensk Kärnbränslehantering AB.

Börgesson L, Johannesson L-E, Hernelind J, 2004. Earthquake induced rock shear through a deposition hole. Effect on the canister and the buffer. SKB TR-2004-02, Svensk Kärnbränslehantering AB.

Börgesson L, Sandén T, Fälth B, Åkesson M, Lindgren E, 2005. Studies of buffer behaviour in KBS-3H concept; Work during 2002–2004. SKB R-05-50, Svensk Kärnbränslehantering AB.

Börgesson L, Hernelind J, 2006. Consequences of loss or missing bentonite in a deposition hole. SKB TR 06-13. Svensk Kärnbränslehantering AB, Sweden.

Carlson L, Karnland O, Olsson S, Rance A, Smart N, 2006. Experimental studies on the interactions between anaerobically corroding iron and bentonite. Posiva Working Report 2006-60 and SKB R-08-28. Posiva Oy, Olkiluoto, Finland and Svensk Kärnbränslehantering AB, Sweden.

Chapman N A, McKinley I G, Penna Franca E, Shea M J, Smellie J A T, 1992. The Poços de Caldas Project. An introduction and summary of its implications for radioactive waste disposal. J. of Geochemical. Exploration, 45, 1–24.

Chapman N A, McCombie C, 2003. Principles and standards for the disposal of long-lived radioactive wastes. Waste Management Series. Vol. 3. Elsevier Science Ltd. Oxford.

Chou N, Martin C D, Christiansson R, 2002. Suppressing fracture growth around underground openings. In Proc. 5th North American Rock Mechanics Symposium and 17th Tunnel Association of Canada Conference NARMS/TAC 2002, Toronto (Ed. Hammah R, Baden W, Curran J and Telesnicki M), vol. 2, pp 1151–1158, University of Toronto Press, Canada.

Cramer J J, Smellie J A T, 1994. The AECL/SKB Cigar Lake Analogue Study. Some implications for performance assessment. In von Maravic, H. and Smellie, J.A.T. (eds) 5th EC NAWG meeting. proceedings of an international workshop held in Toledo, October 1992. CEC Nuclear Science and Technology EUR 15176. CEC, Luxembourg.

Duerden P, (ed) 1990. Alligator Rivers Analogue Project. annual report 1989–1990. ANSTO report. Australian Nuclear Science and Technology Organisation, Lucas Heights, Australia.

Ekström P-A, Broed R, 2006. Sensitivity analysis methods and a biosphere test case implemented in EIKOS. Posiva Working Report 2006-31. Posiva Oy, Olkiluoto, Finland.

Enescu N, Cosma C, Balu L, 2003. Seismic VSP and crosshole investigations in Olkiluoto, 2002. Posiva Working Report 2003-13. Posiva Oy, Olkiluoto, Finland.

Eronen M, Glückert G, van de Plassche O, van de Plicht J, Rantala P, 1995. Land uplift in the Olkiluoto-Pyhäjärvi area, southwestern Finland, during last 8,000 years. Report YJT-95-17. Nuclear Waste Commission of Finnish Power Companies (YJT), Helsinki, Finland.

Freeze R A, Cherry J A, 1979. Groundwater. Prentice Hall, Inc., Englewood Cliffs, USA.

FTRANS, 1983. A two-dimensional code for simulating fluid flow and transport of radioactive nuclides in fractured rock for repository performance assessment. Report ONWI-426. Intera Environmental Consultants Inc., Houston, USA.

Gamsjäger H, Bugajski J, Lemire R, Gajda T, Preis W, 2005. Chemical Thermodynamics 6. Chemical Thermodynamics of Nickel. NEA OECD, Elsevier.

Gribi P, Johnson L, Suter D, Smith P, Pastina B, Snellman M, 2007. Safety assessment for a KBS-3H spent nuclear fuel repository at Olkiluoto – Process Report, POSIVA 2007-09 and SKB R-08-36. Posiva Oy, Olkiluoto, Finland and Svensk Kärnbränslehantering AB, Sweden.

Grivé M, Montoya V, Duro L, 2007. Assessment of the concentration limits for radionuclides for Posiva. Posiva Working Report 2007-103. Posiva Oy, Olkiluoto, Finland.

Haapanen R, Aro L, Ilvesniemi H, Kareinen T, Kirkkala T, Lahdenperä A-M, Mykrä S, Turkki H, Ikonen A T K, 2007. Olkiluoto biosphere description 2006. POSIVA 2007-02. Posiva Oy, Olkiluoto, Finland.

Hagros A, 2007a. Estimated quantities of residual materials in a KBS-3H repository at Olkiluoto. Posiva Working Report 2007-104 and SKB R-08-33. Posiva Oy, Olkiluoto, Finland and Svensk Kärnbränslehantering AB, Sweden.

Hagros A, 2007b. Foreign materials in the repository − Update of estimated quantities. Olkiluoto, Finland. Posiva Working Report 2007-17. Posiva Oy, Olkiluoto, Finland.

Hallberg R O, Ostlund P, Wadsten T, 1987. A 17th century cannon as analogue for radioactive waste disposal. In Come B. and Chapman N.A. (eds.) Natural analogues in radioactive waste disposal. CEC Radioactive Waste Management Series EUR 11037. CEC, Luxembourg.

Hartikainen J, 2006. Numerical simulation of permafrost depth at Olkiluoto. Posiva Working Report 2006-52. Posiva Oy, Olkiluoto, Finland.

Hedin A, 1997. Spent nuclear fuel – how dangerous is it? A report from the project "Description of risk". SKB TR-97-13, Svensk Kärnbränslehantering AB.

Hellä P, Ahokas H, Palmén J, Tammisto E, 2006. Analysis of geohydrological data for design of KBS-3H repository layout. Posiva Working Report 2006-16 and SKB R-08-27. Posiva Oy, Olkiluoto, Finland and Svensk Kärnbränslehantering AB, Sweden.

Hjerpe T, 2006. Overall strategy for handling parameters and data in the Posiva biosphere assessment. Posiva Working Report 2006-109. Posiva Oy, Olkiluoto, Finland.

IAEA (International Atomic Energy Agency), 2003. IAEA Radioactive Waste Management Glossary, 2003 Edition. IAEA, Vienna. Available at <http://www-pub.iaea.org/MTCD/publications/PDF/Pub1155_web.pdf>.

ICRP (International Commission on Radiological Protection), 1991. 1990 Recommendations of the International Commission on Radiological Protection, ICPR Publication 60. Annals of the ICRP Volume 21/1-3. Elsevier Science Ltd, Oxford.

Ikonen K, 2003. Thermal analysis of KBS-3H type repository. POSIVA 2003-11. Posiva Oy, Olkiluoto, Finland.

Ikonen K, 2005. Thermal conditions of open KBS-3H tunnel. POSIVA 2005-04 and SKB R-08-24. Posiva Oy, Olkiluoto, Finland and Svensk Kärnbränslehantering AB, Sweden.

Ikonen A T K, Hjerpe T, Aro L, Leppänen V, 2007. Terrain and ecosystems development model of Olkiluoto site, version 2006. Posiva Working report 2007-110. Posiva Oy, Olkiluoto, Finland.

Johansson E, Äikäs K, Autio J, Hagros A, Malmlund H, Rautakorpi J, Sievänen U, Wanne T, Anttila P, Raiko H, 2002. Preliminary KBS-3H layout adaptation for the Olkiluoto site – Analysis of rock factors affecting the orientation of a KBS-3H deposition hole. Posiva Working Report 2002-57. Posiva Oy, Olkiluoto, Finland.

Johansson E, Hagros A, Autio J, Kirkkomäki T, 2007. KBS-3H layout adaptation 2007 for the Olkiluoto site. Posiva Working Report 2007-77 and SKB R-08-31. Posiva Oy, Olkiluoto, Finland and Svensk Kärnbränslehantering AB, Sweden.

Johnson L H, Tait J C, 1997. Release of segregated radionuclides from spent fuel. SKB TR-97-18, Svensk Kärnbränslehantering AB.

Johnson L, Marschall P, Wersin P, Gribi P, 2005. HMCBG processes related to the steel components in the KBS-3H disposal concept. Posiva Working Report 2005-09 and SKB R-08-25. Posiva Oy, Olkiluoto, Finland and Svensk Kärnbränslehantering AB, Sweden.

Kahma K, Johansson M, Boman H, 2001. Meriveden pinnankorkeuden jakauma Loviisan ja Olkiluodon rannikoilla seuraavien 30 vuoden aikana. Merentutkimuslaitos PGN, Helsinki. 25 pp.

King F, 2002. Corrosion of copper in alkaline chloride environments. SKB TR-02-25, Svensk Kärnbränslehantering AB.

Lahdenperä A-M, Palmén J, Hellä P, 2005. Summary of overburden studies at Olkiluoto with an emphasis on geosphere-biosphere interface. Posiva Working Report 2005-11. Posiva Oy, Olkiluoto, Finland.

Lambeck K, Smither C, Johnston P, 1998. Sea-level change, glacial rebound and mantle viscosity for northern Europe. Geophys. J. Int., 134, 102–144.

Lanyon G W, Marschall P, 2006. Discrete fracture network modelling of a KBS-3H repository at Olkiluoto. POSIVA 2006-06 and SKB R-08-26. Posiva Oy, Olkiluoto, Finland and Svensk Kärnbränslehantering AB, Sweden.

La Pointe P, Hermanson J, 2002. Estimation of rock movements due to future earthquakes at four candidate sites for a spent fuel repository in Finland. POSIVA 2002-02. Posiva Oy, Olkiluoto, Finland.

Lindgren M, Lindström F, 1999. Radionuclide transport calculations. SKB TR-99-23, Svensk Kärnbränslehantering AB.

Löfman J, 1996. Groundater flow modelling at the Olkiluoto site– Flow under natural conditions. Posiva Working Report PATU-96-76e. Posiva Oy, Helsinki, Finland.

Löfman J, 2005. Simulation of hydraulic disturbances caused by decay heat of the repository in Olkiluoto. POSIVA 2005-07. Posiva Oy, Olkiluoto, Finland.

Lönnqvist M, Hökmark H, 2007. Thermo-mechanical analyses of a KBS-3H deposition drift at the Olkiluoto site, Posiva Working Report 2007-66 and SKB R-08-30. Posiva Oy, Olkiluoto, Finland and Svensk Kärnbränslehantering AB, Sweden.

Marcos N, 1989. Native copper as a natural analogue for copper canisters. Report YJT-89-18. Nuclear Waste Commission of Finnish Power Companies. Helsinki, Finland.

Marcos N, Hellä P, Snellman M, Nykyri M, Pastina B, Vähänen M, Koskinen K, 2007. The Role of the Geosphere in Posiva's Safety Case. Paper presented at the IGSC "Geosphere Stability" Workshop on the Stability and Buffering Capacity of the Geosphere for Long-term

Isolation of Radioactive Waste: Application to Crystalline Rock. Manchester, 13–15 November 2007.

Munier R, 2006. Using observations in deposition tunnels to avoid intersections with critical features in deposition holes. SKB R-06-54, Svensk Kärnbränslehantering AB.

Müller Ch, Elagin M, Scharmach M, Bellon C, Jaenisch G-R, Bär S, Redmer B, Goebbels J, Ewert U, Zscherpel U, Boehm R, Brekow G, Erhard A, Heckel T, Tessaro U, Tscharntke D, Ronneteg U, 2006. Reliability of nondestructive testing (NDT) of the copper canister seal weld. SKB R-06-08, Svensk Kärnbränslehantering AB.

Mäkiaho J-P, 2005. Development of Shoreline and Topography in the Olkiluoto Area, Western Finland, 2000 BP – 8000 AP. Posiva Working Report 2005-70. Posiva Oy, Olkiluoto, Finland.

Nagra, 2002. Project Opalinus Clay: Safety Report. Demonstration of disposal feasibility for spent fuel, vitrified high-level waste and long-lived intermediate-level waste (Entsorgungsnachweis). Nagra Technical Report NTB 02-05. Nagra, Wettingen, Switzerland.

NEA, 1997. Lessons learnt from ten performance assessments studies. A Report of the NEA Working Group on Intergrated Performance Assessments of Deep Repositories. OECD/NEA, Paris, France.

NEA, 2000. Features, Events and Processes (FEPs) for geologic disposal of radioactive waste: an international database, OECD, Paris 2000.

Neall F, Pastina B, Smith P, Gribi P, Snellman M, Johnson L, 2007. KBS-3H Complementary evaluations of safety. POSIVA 2007-10. and SKB R-08-35 Posiva Oy, Olkiluoto, Finland. and Svensk Kärnbränslehantering AB, Sweden.

Nordman H, Vieno T, 1994. Near-field model REPCOM. Report YJT-94-12. Nuclear Waste Commission of Finnish Power Companies. (YJT), Helsinki, Finland.

Nordman H, Vieno T, 2003. Modelling of near-field transport in KBS-3V/H type repositories with PORFLOW and REPCOM codes. Posiva Working Report 2003-07. Posiva Oy, Olkiluoto, Finland.

Ojala A E K, Virkki H, Palmu J-P, Hokkanen K, Kaija J, 2006. Regional development of river basins in the Olkiluoto-Pyhäjärvi area, SW Finland, 2000 BP – 8000 AP. Posiva Working report 2006-113. Posiva Oy, Olkiluoto, Finland.

Pastina B, Hellä P, 2006. Expected evolution of a spent fuel repository at Olkiluoto. POSIVA 2006-05. Posiva Oy, Olkiluoto, Finland.

Pitkänen P, Partamies S, 2007. Origin and implications of dissolved gases in groundwater at Olkiluoto. POSIVA 2007-04. Posiva Oy, Olkiluoto, Finland.

Pitkänen P, Partamies S, Luukkonen A, 2004. Hydrogeochemical interpretation of baseline groundwater conditions at the Olkiluoto site. POSIVA 2003-07. Posiva Oy, Olkiluoto, Finland.

Posiva, 2003. Baseline conditions at Olkiluoto. POSIVA 2003-02. Posiva Oy, Olkiluoto, Finland.

Posiva, 2005. Olkiluoto Site Description 2004. POSIVA 2005-03. Posiva Oy, Olkiluoto, Finland.

Posiva, 2006. Nuclear waste management of the Olkiluoto and Loviisa power plants: Programme for research, development and technical design for 2007-2009. Posiva TKS-2006. Posiva Oy, Olkiluoto, Finland.

Påsse T, 1996. A mathematical model of shore level displacement in Fennoscandia. SKB TR-96-24, Svensk Kärnbränslehantering AB.

Påsse T, 1997. A mathematical model of past, present and future shore level displacement in Fennoscandia. SKB TR-97-28, Svensk Kärnbränslehantering AB.

Påsse T, 1998. Lake-tilting, a method for estimation of glacio-isostatic uplift. *Boreas* 27:1, 69-80.

Raiko E, Nordman H, 1999. Kemiallinen myrkyllisyys käytetyn ydinpolttoaineen loppusijoituksessa. Posiva Working Report WR-99-18 (in Finnish with English abstract***)***. Partial translation by K-H Hellmuth given in: Pitkänen P, Löfman J, Luukkonen A and Partamies S, 2003. Chemical toxicity in final disposal of spent nuclear fuel. IAEA Coordinated Research Project. The use of selected safety indicators (concentrations; fluxes) in the assessment of radioactive waste disposal. Report 7. Site-specific natural geochemical concentrations and fluxes at four repository investigation sites in Finland as use for indicators of nuclear waste repository safety. Supplement 1. GTK Report YST-115. Geological Survey of Finland.

Raiko H, 2005. Disposal canister for spent nuclear fuel – Design Report. POSIVA 2005-02. Posiva Oy, Olkiluoto, Finland.

Rasilainen K, (ed), 2004. Localisation of the SR 97 process report for Posiva's spent fuel repository at Olkiluoto. POSIVA 2004-05. Posiva Oy, Olkiluoto, Finland.

Ronneteg U, Cederqvist L, Rydén H, Öberg T, Müller C, 2006. Reliability in sealing of canister for spent nuclear fuel. SKB R-06-26, Svensk Kärnbränslehantering AB.

Ruokola E, 2002. Consideration of timescales in the Finnish safety regulations for spent fuel disposal. Proceedings of Workshop on the handling of timescales in assessing post-closure safety of deep geological repositories. Organisation for Economic Co-operation and Development, Nuclear Energy Agency, p. 85–89.

Ruosteenoja K, 2003. Climate scenarios for Olkiluoto over the next few centuries, Posiva Working Report 2003-21. Posiva Oy, Olkiluoto, Finland.

Saanio T, Kirkkomäki T, Sacklén N, Autio J, Kukkola T, Raiko H, 2004. Spent nuclear fuel repository at Olkiluoto – Preliminary design – Stage 1. Posiva Working Report 2003-74 (In Finnish). Posiva Oy, Olkiluoto, Finland.

Saanio T, Kirkkomäki T, Keto P, Kukkola T, Raiko H, 2006. Preliminary design of the repository, Stage 2. Posiva Working Report 2006-94. Posiva Oy, Olkiluoto, Finland.

Saari J, 2006. Local seismic network at the Olkiluoto Site – Annual Report for 2005. Posiva Working Report 2006-57. Posiva Oy, Olkiluoto, Finland.

Sandén T, Börgesson L, Dueck A, Goudarzi R, Lönnqvist M, Nilsson U, Åkesson M, 2008. Description of buffer tests in 2005–2007. KBS-3H. Results from laboratory tests. Clay Technology. SKB R-08-40 and Posiva Working Report 2008-10 (in preparation). Svensk Kärnbränslehantering AB, Sweden and Posiva Posiva Oy, Olkiluoto, Finland.

SKB, 1999a. Deep repository for spent nuclear fuel, SR 97 – Post closure safety. SKB TR-99-06, Svensk Kärnbränslehantering AB.

SKB, 1999b. Waste, repository design and sites. SKB TR-99-08, Svensk Kärnbränslehantering AB.

SKB, 1999c. SR 97 – Processes in the repository evolution. SKB TR-99-07, Svensk Kärnbränslehantering AB.

SKB, 2006a. Long-term safety for KBS-3 repositories at Forsmark and Laxemar – a first evaluation – Main report of the SR-Can project. SKB TR-06-09, Svensk Kärnbränslehantering AB.

SKB, 2006b. Data report for the safety assessment SR-Can. SKB TR-06-25, Svensk Kärnbränslehantering AB.

SKB, 2006c. Model summary report for the safety assessment SR-Can. SKB TR-06-26, Svensk Kärnbränslehantering AB.

SKB 2006d. Fuel and canister process report for the safety assessment SR-Can. SKB TR-06-22, Svensk Kärnbränslehantering AB.

SKB, 2007. Fud-program 2007. Program för forskning, utveckling och demonstration av metoder för hantering och slutförvaring av kärnavfall. Svensk Kärnbränslehantering AB, pages 173–176.

SKB/Posiva, 2008. Horizontal deposition of canisters for spent nuclear fuel. Summary report of the KBS-3H Project 2004–2007. SKB Technical Report TR-08-03. POSIVA 2008-03 (in preparation). Svensk Kärnbränslehantering AB, Sweden and Posiva Oy, Olkiluoto, Finland.

Smith P, Nordman H, Pastina B, Snellman M, Hjerpe T, Johnson L, 2007a. Safety assessment for a KBS-3H spent nuclear fuel repository at Olkiluoto – Radionuclide transport report. POSIVA 2007-07 and SKB R-08-38. Posiva Oy, Olkiluoto, Finland and Svensk Kärnbränslehantering AB, Sweden.

Smith P A, Johnson L H, Snellman M, Pastina B, Gribi P, 2007b. Safety assessment for a KBS-3H spent nuclear fuel repository at Olkiluoto – Evolution report. POSIVA 2007-08 and SKB R-08-37. Posiva Oy, Olkiluoto, Finland and Svensk Kärnbränslehantering AB, Sweden.

STUK, 1999. Government decision on the safety of disposal of spent nuclear fuel (478/1999). Radiation and Nuclear Safety Authority (STUK) Report STUK-B-YTO 195. STUK, Helsinki, Finland.

STUK, 2001. Long-term safety of disposal of spent nuclear fuel. Radiation and Nuclear Safety Authority (STUK). Guide YVL 8.4. STUK, Helsinki, Finland.

Vieno T, Nordman H, 1999. Safety assessment of spent fuel disposal in Hästholmen, Kivetty, Olkiluoto and Romuvaara TILA-99. POSIVA 99-07. Posiva Oy, Helsinki, Finland.

Vieno T, 2000. Groundwater salinity at Olkiluoto and its effects on a spent fuel repository. POSIVA 2000-11. Posiva Oy, Helsinki, Finland.

Vieno T, Lehikoinen J, Löfman J, Nordman H, Mészáros F, 2003. Assessment of disturbances caused by construction and operation of ONKALO. POSIVA 2003-06. Posiva Oy, Helsinki, Finland.

Vieno T, Ikonen A T K, 2005. Plan for Safety Case of spent fuel repository at Olkiluoto. POSIVA 2005-01. Posiva Oy, Olkiluoto, Finland.

Werme L O, Johnson L H, Oversby V M, King F, Spahiu K, Grambow B, Shoesmith DW, 2004. Spent fuel performance under repository conditions: A model for use in SR-Can. SKB TR 04-19, Svensk Kärnbränslehantering AB.

Wersin P, Birgersson M, Olsson S, Karnland O, Snellman M, 2007. Impact of corrosion-derived iron on the bentonite buffer within the KBS-3H disposal concept. The Olkiluoto site as case study. POSIVA 2007-11 and SKB R-08-34. Posiva Oy, Olkiluoto, Finland and Svensk Kärnbränslehantering AB, Sweden.

Åstrand P-G, Jones J, Broed R, Avila R, 2005. PANDORA technical description and user guide. Posiva Working Report 2005-64. Posiva Oy, Olkiluoto, Finland.

Appendix

Glossary

Air evacuation	Removing of air from a drift compartment through pipes during artificial watering.
Artificial watering	Adding water through pipes to a supercontainer section to facilitate buffer saturation.
Backfilling	Filling the deposition niche, transport tunnels and other parts of the repository.
Basic Design	KBS-3H design alternative.
BD	Basic Design.
Buffer	Bentonite originally inside the supercontainers and the bentonite distance blocks.
Candidate design	Design alternative to be used for selecting a suitable design.
Catching tube	Equipment for catching the copper canister during retrieval.
Compartment	Drift section used for emplacement of supercontainers. Typically, the 300 m-long drift is divided into 2 compartments by a compartment plug.
Compartment plug	Steel plug used to seal off drift sections where inflows are higher than 1 litre per minute after grouting, thus dividing the drift into compartments.
Cutting tool	Device for removal/cutting of supercontainer end plate during retrieval.
DAWE	Drainage, Artificial Watering and air Evacuation design alternative. KBS-3H design alternative.
DD-2005, DD-2006, DD-2007	KBS-3H Design Description 2005, 2006 and 2007 reports, respectively.
Deposition drift	100–300 m long hole with a diameter of 1.85 m for horizontal emplacement of supercontainers.
Deposition equipment	Includes all equipment needed for the emplacement of supercontainer and installation of distance blocks.
Deposition machine	The machine used in the deposition drift for emplacement of supercontainers and distance blocks.
Deposition niche	A tunnel section in front of the deposition drift hosting the deposition equipment.
Design component	A component in design which fulfils a specific functional requirement, e.g. compartment plug, distance blocks.
Distance blocks	Bentonite blocks between the supercontainers. The roles of the distance blocks are to provide hydraulic separation and thermal spacing.

Drift end plug	A steel-reinforced low-pH concrete bulkhead positioned in a notch situated at the end of deposition drift close to the intersection with the deposition niche.
Drip (and spray) shield	Thin steel (or copper) sheets over inflow points preventing erosion of bentonite due to the spraying, dripping and squirting of water from the drift walls onto the distance blocks and supercontainers.
EDZ	Excavation Damaged Zone; section of the rock damaged by the boring of deposition drifts.
End plate	Unperforated steel end plate for the supercontainer shell.
Engineered and residual materials	Materials introduced during construction and operation of the repository that will remain underground after closure.
Erosion	Loss or redistribution of bentonite mass in the deposition drift due to physical or chemical processes, such as piping or chemical erosion by dilute water.
Fastening ring	Steel ring used to fasten the steel compartment plug to the rock.
Filling block	Filling blocks are placed at positions where supercontainer units cannot be positioned because inflow is higher than positioning criteria.
Filling material	Material between and in the vicinity of the compartment plugs to fill empty space which cannot be filled by using filling blocks.
Fixing ring (BD design only)	Steel rings installed, where necessary, to avoid displacement of the distance blocks prior to the installation of compartment and drift end plugs.
Gamma gate	Sliding radiation protection gates located on the transport tube or at the entrance of the deposition drift.
Gripping tool	Device for removal of canister from the drift during retrieval.
Handling cell	Shielded space for handling of spent fuel canister.
Handling equipment	Equipment for handling of spent fuel canister within the reloading station.
Horizontal push-reaming	Excavation method to ream the pilot hole to full drift size, known also as horizontal blindboring, reverse raiseboring or horizontal box-hole boring.
KBS	(**K**ärn**b**ränsle**s**äkerhet). The method for implementing the spent fuel disposal concept based on multiple barriers (as required in Sweden and in Finland). KBS-1, KBS-2 and KBS-3 are variations of this method.
KBS-3H	(**K**ärn**b**ränsle**s**äkerhet **3-H**orisontell). Design alternative of the KBS-3 method in which several spent fuel canisters are emplaced horizontally in each deposition drift.

KBS-3V	(**K**ärn**b**ränsle**s**äkerhet **3-V**ertikal). The reference design alternative of the KBS-3 method in which the spent fuel canisters are emplaced in individual vertical deposition holes.
LHHP cement	Low-Heat High-Performance cement, used for spent fuel repository applications, characterized by a low heat of hydration, and a lower release of free hydroxide ions and lower pH than for ordinary cement.
Mega-Packer	Large-scale post-grouting device for grouting of rock.
ONKALO	Underground rock characterisation facility in Olkiluoto, Finland.
Parking feet	Feet on the supercontainer
Pilot hole	Rotary drilled hole for guiding horizontal push-reaming excavation.
Piping	Formation of hydraulically conductive channels in the bentonite due too high water flow and hydraulic pressure difference along the drift.
Post-grouting	Grouting method used in deposition drift after excavation.
Pre-grouting	Grouting made through investigation or pilot holes before reaming the drift to full size.
Pre-pilot hole	Core-drilled investigation hole made before drilling the pilot hole. This may be used for guiding they boring of pilot hole.
Reloading station	Station at repository level where the spent fuel canister is transferred from the transport cask to the supercontainer.
Retrievability	Possibility of removal of canisters after the buffer has absorbed water and started to swell within the deposition drift.
Retrieval	Removal of the canister after the buffer has absorbed water and started to swell within the deposition drift.
Reverse operation	Operation to remove the supercontainer from the deposition drift before the buffer has absorbed water and started to swell within the deposition drift.
Safety studies	Long-term safety studies performed for the 2004–2007 KBS-3H project consisting of five main reports: Process, Evolution, Radionuclide transport, Complementary Evaluations of Safety and Summary.
Sealing ring	Design component in the STC design (still at a conceptual stage) presented in DD-2007.
Silica Sol	Type of colloidal silica used for groundwater control purposes.
Spalling	Breaking of the rock surface of deposition drift induced by high rock stresses into splinters, chips or fragments.
Start tube	Support structure for the deposition machine.
STC	Semi Tight Compartments design alternative.

Supercontainer	Assembly consisting of a canister surrounded by bentonite clay and a perforated shell.
Supercontainer section	Section of the drift (about 10 m long for the reference type BWR fuel from Olkiluoto 1-2) in which a supercontainer and a distance block are located.
Supercontainer shell	Perforated shell (8 mm thick) that holds together the canister and the bentonite surrounding it.
Transition block	A component in the filling system adjacent to compartment plug.
Transport tube	Tube for the handling of the supercontainer.
Transport vehicle	Vehicle for transportation of deposition equipment.
Water cushion system	System for the transportation of supercontainers and distance blocks.
Äspö HRL	Äspö Hard Rock Laboratory, near Oskarshamn, Sweden.